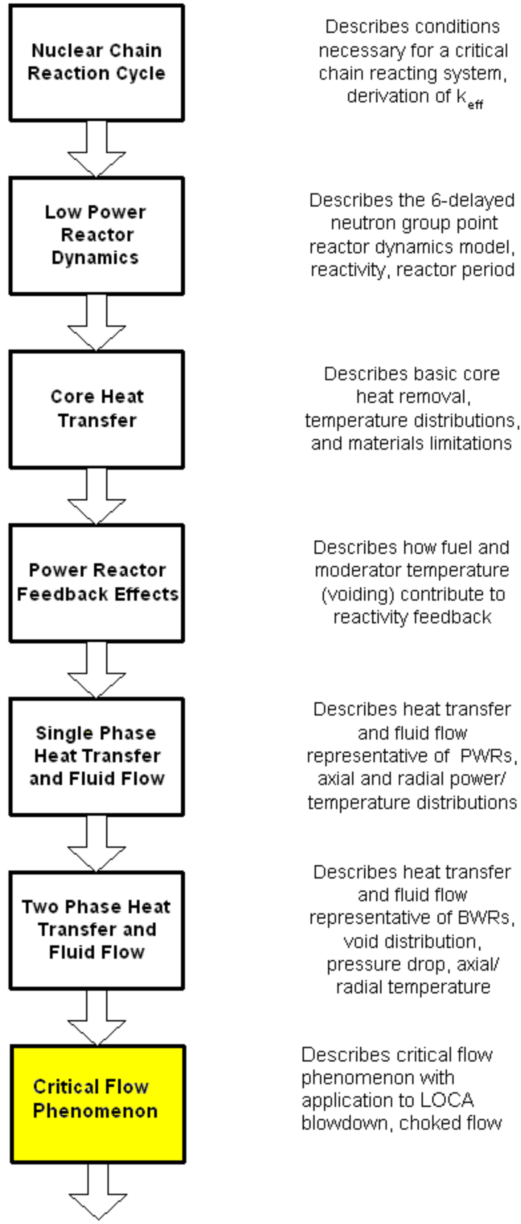
Fundamentals of Nuclear Engineering

Module 13: Critical Flow Phenomenon

Joseph S. Miller, P.E.


Nuclear Chain Reaction Cycle
Describes conditions necessary for a critical chain reacting system, derivation of k_{eff}
Low Power Reactor Dynamics
Describes the 6-delayed neutron group point reactor dynamics model, reactivity, reactor period
Core Heat Transfer
Describes basic core heat removal, temperature distributions, and materials limitations
Power Reactor Feedback Effects
Describes how fuel and moderator temperature (voiding) contribute to reactivity feedback
Single Phase Heat Transfer and Fluid Flow
Describes heat transfer and fluid flow representative of PWRs, axial and radial power/ temperature distributions
Two Phase Heat Transfer and Fluid Flow
Describes heat transfer and fluid flow representative of BWRs, void distribution, pressure drop, axial/ radial temperature
Critical Flow Phenomenon
Describes critical flow phenomenon with application to LOCA blowdown, choked flow

Objectives:

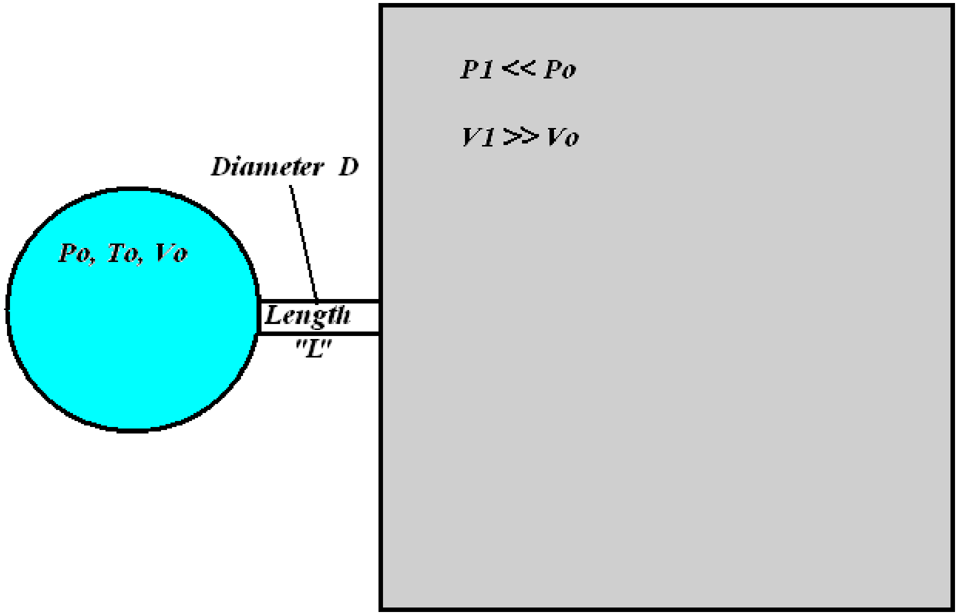
Previous Lectures described single and two-phase fluid flow in various systems. This lecture:

1. Describe Critical Flow – What is it
2. Describe Single Phase Critical Flow
3. Describe Two-Phase Critical Flow
4. Describe Situations Where Critical Flow is Important
5. Describe origins and use of Some Critical Flow Correlations
6. Describe Some Testing that has been Performed for break flow and system performance

Critical Flow Phenomenon

1. What is Critical Flow?

- Envision 2 volumes at different pressures suddenly connected
- Critical flow ("*choked flow*") involves situation where fluid moves from higher pressure volume at speed limited only by *speed of sound* for fluid – such as LOCA
- Various models exist to describe this limiting flow rate:
- One-phase vapor, one phase liquid, subcooled flashing liquid, saturated flashing liquid, and two-phase conditions



2. Critical Single Phase Flow

Three Examples will be given

1. Steam Flow
2. Ideal Gas
3. Incompressible Liquid

Critical Single Phase Flow - Steam

- In single phase flow: sonic velocity a and critical mass flow are directly related:

$$a^2 = \frac{g_c v(P,T)^2}{\left(\frac{dv(P,T)}{dP}\right)_S}$$

$$G_{crit}{}^2 = a^2 \rho(P,T)^2 = \frac{g_c}{\left(\frac{dv(P,T)}{dP}\right)_S}$$

- Derivative term is *total derivative of specific volume* evaluated at *constant entropy*
- Tabulated values of critical steam flow can be found in steam tables

Example Critical Steam Flow Calculation

- Assume 2 in^2 steam relief valve opens at 1000 psi
- What is steam mass discharge rate?
- Assume saturated system with $T_{sat} = 544.61°F$
- $f = 50.3$
- $W_{crit} = f\,P\,A$

  $=(50.3\text{lb-m/hr})(1000\text{ psi})(2\text{ in}^2)$

  $= 100{,}600\text{ lb-m/hr}$

  $= 27.94\text{ lb-m/sec.}$

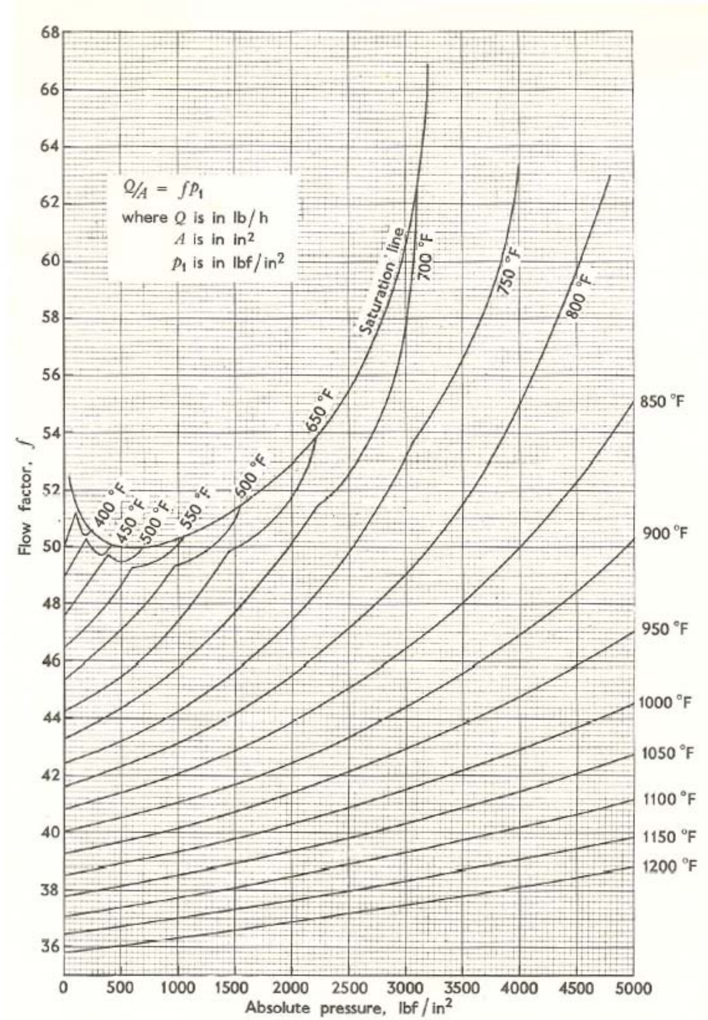


Fig. 5. Critical mass flow per unit area as a function of initial pressure and temperature.

Mass flow rate of a gas at choked conditions for Ideal Gas

All gases flow from upstream higher stagnation pressure sources to downstream lower pressure sources. There are several situations in which choked flow occurs, such as: change of cross section (as in a convergent-divergent nozzle or flow through an orifice plate.

When the gas velocity is choked, the equation for the mass flow rate in SI metric units is:

$$m = C\,A\sqrt{k\,\rho\,P\left(\frac{2}{k+1}\right)^{(k+1)/(k-1)}}$$

or for an ideal gas

$$m = C\,A\,P\sqrt{\left(\frac{k\,M}{Z\,R\,T}\right)\left(\frac{2}{k+1}\right)^{(k+1)/(k-1)}}$$

where:

m= mass flow rate, kg/s

C = discharge coefficient, dimensionless (usually about 0.72)

A = discharge hole cross-sectional area, m²

k= c_p/c_v of the gas

c_p = specific heat of the gas at constant pressure

c_v = specific heat of the gas at constant volume

ρ = real gas density at P and T, kg/m³

P= absolute upstream stagnation pressure, Pa

M= the gas molecular mass, kg/kmole (also known as the molecular weight)

R= Universal gas law constant = 8314.5 (N·m) / (kmole·K)

T= absolute gas temperature, K

Z= the gas compressibility factor at P and T, dimensionless

Critical Single Phase Flow – Incompressible Liquid (Lahey and Moody, Ref. 1)

When a flow passage opens between the reactor coolant system and its environment by either a pipe rupture or some other mechanism, fluid is expelled at a blowdown rate, w. If the blowdown passage transfers no energy to or from the fluid,

$$h_0 = \langle h \rangle + \frac{\langle u \rangle^2}{2g_c J} = \text{constant}\,, \tag{9.32}$$

where h_0 is the stagnation or total enthalpy. Moreover, for ideal reversible flow, the Second Law of Thermodynamics and the Gibbs equation yield,

$$d\langle h \rangle = \frac{dp}{J\langle \rho \rangle}\,. \tag{9.33}$$

By assuming a one-dimensional velocity profile and by introducing the definition of mass flux, $G = \langle \rho \rangle \langle u \rangle$, Eq. (9.33) can be integrated between the stagnation state in the vessel and properties at the throat of an ideal nozzle. By using Eqs. (9.32) and (9.33), the resulting ideal mass flux (G_i) is,

$$G_i = \langle \rho \rangle [2g_c J(h_0 - \langle h \rangle)]^{1/2} = \langle \rho \rangle \left(2g_c \int_{p_t}^{p_0} \frac{dp}{\langle \rho \rangle} \right)^{1/2}. \tag{9.34}$$

For incompressible liquid flow, $\langle \rho \rangle = \rho_l$. Thus, Eq. (9.34) becomes,

$$G_{l_i} = [2g_c \rho_l p_0 (1 - p_t/p_0)]^{1/2}\,, \tag{9.35}$$

Where throat pressure, pr, is equal to the downstream receiver pressure, pR, and Gl, is the so-called Bernoulli mass flux.

Critical Single Phase Flow – Ideal Gas(Lahey and Moody, Ref. 1) (cont)

For compressible gas flow in an ideal nozzle, local pressure and density are normally characterized by the isentropic relationship,

$$p/\rho_g{}^K = p_0/\rho_{g0}{}^K \,. \tag{9.36}$$

By substituting $\langle\rho\rangle = \rho_g$ into Eq. (9.34), it follows that,

$$G_{g_i} = \left\{2g_c\left(\frac{K}{K-1}\right)p_0\rho_{g0}\left[\left(\frac{p_t}{p_0}\right)^{\frac{2}{K}} - \left(\frac{p_t}{p_0}\right)^{\frac{K+1}{K}}\right]\right\}^{1/2} . \tag{9.37}$$

The flow of incompressible liquid and ideal gas have unique differences. Consider a nozzle discharging into a receiver at pressure, p_R. Before the flow rate can be predicted, it is necessary to obtain throat pressure, p_t. Whenever the discharge flow is subsonic, receiver pressure readily propagates to the throat so that,

$$p_t = p_R \text{ (subsonic)} \,. \tag{9.38}$$

Equation (9.38) always applies for incompressible liquid flows and, as seen in Fig. 9-9, G_{l_i} continuously increases with decreasing p_t. However, as also shown in Fig. 9-9, if receiver pressure is decreased for the flow of an ideal compressible gas, a critical condition is reached at which the flow rate reaches a maximum. For this condition, the corresponding discharge is sonic. Therefore, receiver pressure cannot propagate to the throat and further decrease of p_R does not change the value of flow rate or throat pressure. That is,

$$p_t \geq p_R \text{ (sonic)} \,. \tag{9.39}$$

The mathematical maximum value of G_{g_i} is determined by the condition at the throat,

$$\frac{dG_{g_i}}{dp_t} = 0 \,. \tag{9.40}$$

It can be easily shown from Eq. (9.37) that the condition of Eq. (9.40) leads to the so-called critical pressure ratio,

$$\frac{p_t}{p_0} \triangleq \frac{p_c}{p_0} = \left(\frac{2}{K+1}\right)^{\frac{K}{K-1}} . \tag{9.41}$$

Ideal Gas Flow Rate

By combining Eqs. (9.37) and (9.41), the corresponding critical mass flux for an ideal gas is,

$$G_{g_c} = \left[K g_c p_0 \rho_{g0} \left(\frac{2}{K+1} \right)^{\frac{K+1}{K-1}} \right]^{1/2} . \tag{9.42}$$

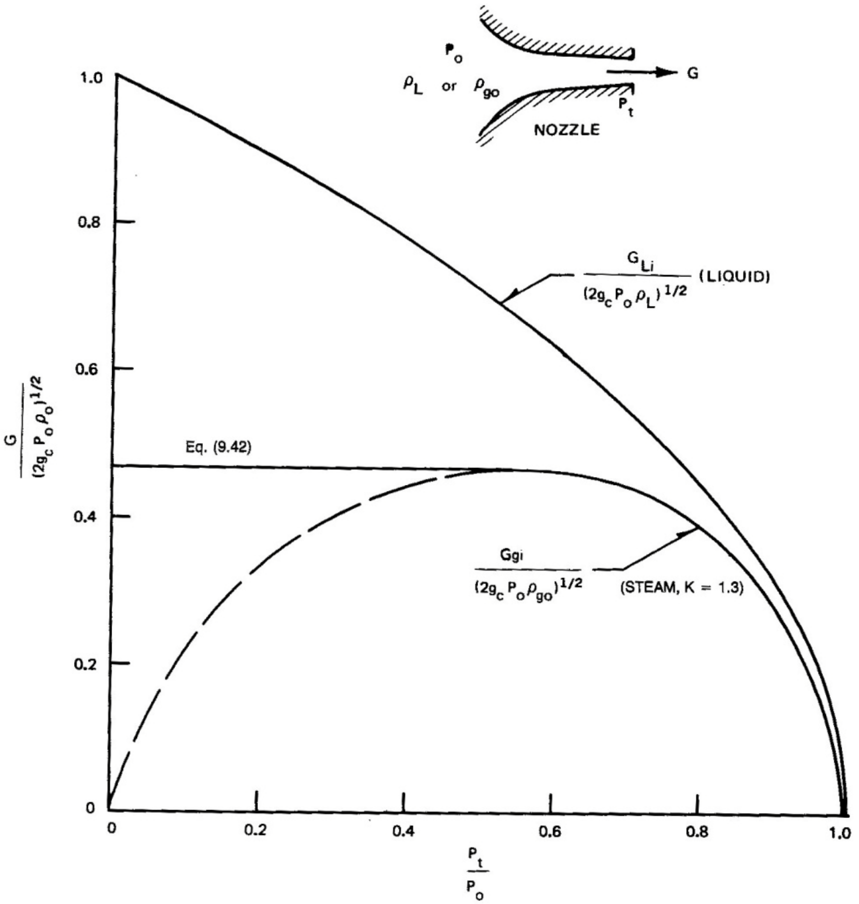


Fig. 9-9. Ideal nozzle flow rates.

3. Two-phase Critical Flow Using Moody Equilibrium Flow Model (Refs. 1 & 2)

The following is a derivation of a thermodynamic equilibrium critical flow model that has come to be known as the Moody model (Moody, 1965). Consider a two-phase system in which unequal phase velocities occur. For this situation, Eq. (9.32) becomes,

$$h_0 = \langle h \rangle + \langle x \rangle \frac{\langle u_g \rangle_g^2}{2g_c J} + (1 - \langle x \rangle) \frac{\langle u_f \rangle_f^2}{2g_c J}, \tag{9.43}$$

where the two-phase enthalpy of the saturated mixture is given by,

$$\langle h \rangle \triangleq h_f + \langle x \rangle h_{fg} . \tag{9.44}$$

If the flow is approximated by an isentropic process,

$$s_f + \langle x \rangle s_{fg} = \text{constant} \triangleq s_0 . \tag{9.45}$$

By eliminating $\langle x \rangle$ between Eqs. (9.44) and (9.45), we obtain a saturation state equation in the form,

$$\langle h \rangle = h_f + \frac{h_{fg}}{s_{fg}} (s_0 - s_f) = \langle h(p, s_0) \rangle . \tag{9.46}$$

Furthermore, a functional relationship exists among stagnation properties in the form,

$$s_0 = s_0(p_0, h_0) . \tag{9.47}$$

Equations (9.43), (9.46), (9.47), (5.18), and (5.19) yield

$$G = \langle \rho''' \rangle \{2g_c J[h_0 - \langle h(p, p_o, h_o) \rangle]\}^{1/2} , \tag{9.48}$$

where the so-called energy density, $\langle \rho''' \rangle$, defined in Eq. (5.88), can be rewritten in terms of the slip ratio, S, as,

$$\langle \rho''' \rangle = \left\{ \left[\frac{\langle x \rangle}{\rho_g} + \frac{S(1 - \langle x \rangle)}{\rho_f} \right] \left[\langle x \rangle + \frac{(1 - \langle x \rangle)}{S^2} \right]^{1/2} \right\}^{-1} . \tag{9.49}$$

Two-phase Critical Flow Using Moody Equilibrium Flow Model (Refs. 1 & 2)

Note that Eq. (9.48) reduces to the Bernoulli mass flux of Eq. (9.35) when $\langle x \rangle \rightarrow 0$. It is readily seen from Eqs. (9.45) through (9.49) that for given stagnation properties, p_0 and h_0,

$$G = G(p, S; p_0, h_0) \, . \tag{9.50}$$

From Eq. (9.37), G_{g_i} is seen to be a function only of throat pressure for a given stagnation state, so that the total derivative of Eq. (9.40) is appropriate. However, for *the two-phase system*, Eq. (9.50) expresses G as a function of *both local static* pressure, p, and the slip ratio, S, for a given stagnation state. If both p and S are considered independent, a maximum G corresponds to the conditions,

$$\left(\frac{\partial G}{\partial S}\right)_p = 0; \left(\frac{\partial^2 G}{\partial S^2}\right)_p < 0 \, , \tag{9.51}$$

and,

$$\left(\frac{\partial G}{\partial p}\right)_S = 0; \left(\frac{\partial^2 G}{\partial p^2}\right)_S < 0 \, . \tag{9.52}$$

Two-phase Critical Flow Using Moody Equilibrium Flow Model (Refs. 1 & 2) (cont)

That is, for a known stagnation state, local static pressure, p, and slip ratio, S, the flow rate unit area is uniquely determined. From numerous experimental and theoretical studies, the slip ratio, S, has been determined to range from S = 1.0 (homogeneous) to $S = (\rho_f / \rho_g)^{1/2}$.

For condition imposed by Moody, the slip ratio is evaluated as

$$S = S(p) = (\rho_f / \rho_g)^{1/3} . \qquad (9.53)$$

Equation (9.53) shows that at a maximum G, the slip ration is a function of pressure only. Employing Eq. (9.53) in Eq. (9.52) and using saturated steam-water properties, Fig. 9-10a can be obtained, which gives G_c in terms of p_0 and h_0 .

Two-phase Critical Flow Using Moody Equilibrium Flow Model (Refs. 1 & 2) (cont)

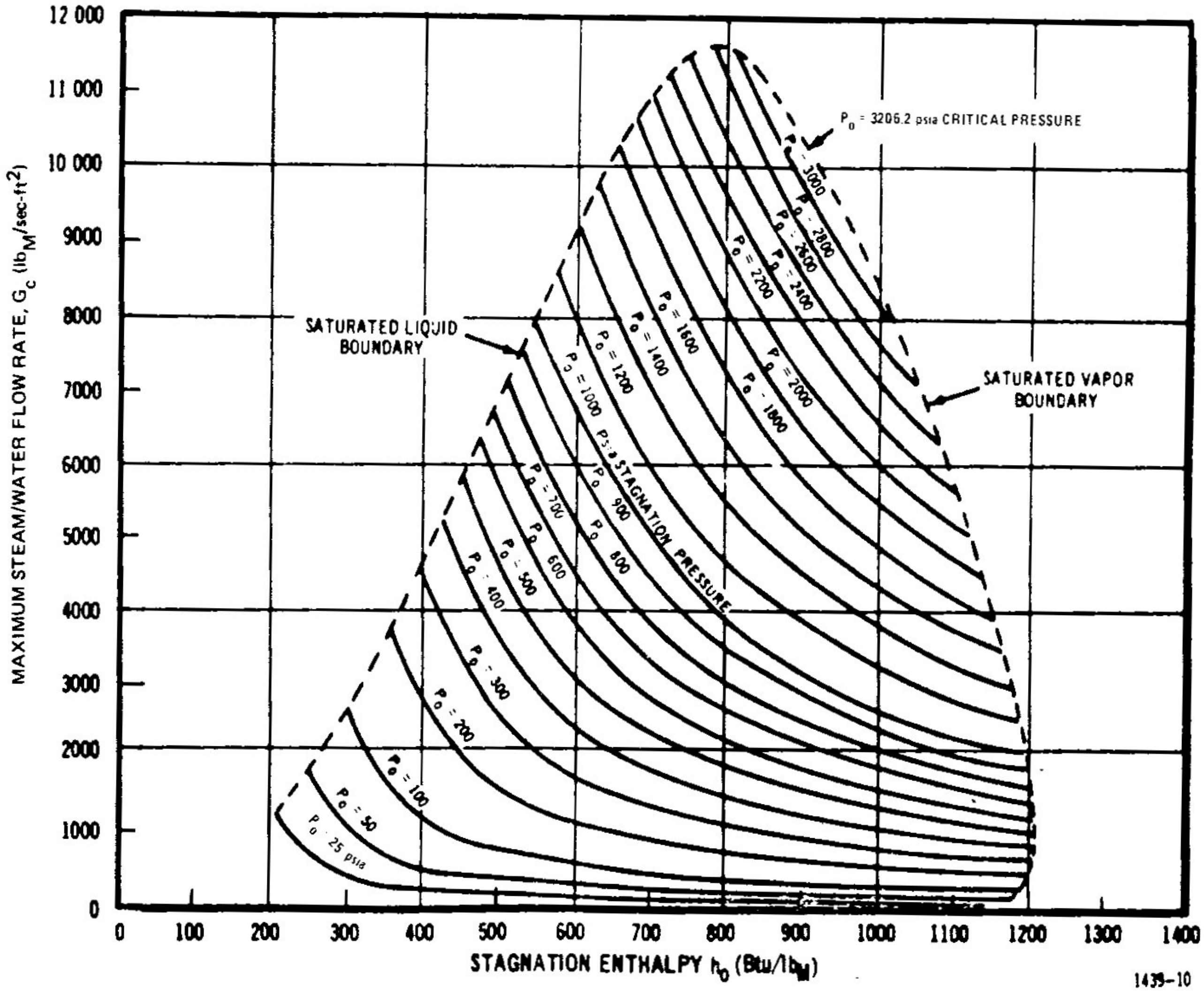


Fig. 9-10a. Maximum steam/water flow rate and local stagnation properties (Moody model).

Example Calculation

From page 8 for sat steam at 1000 psia:

$$W_{crit} = f\,P\,A$$

$$= (50.3 \text{ lb-m/hr})(1000 \text{ psi})(2 \text{ in}^2)$$

$$= 100{,}600 \text{ lb-m/hr}$$

$$= 27.94 \text{ lb-m/sec.}$$

What is this value when using Moody table on page 16?

$G_c(P = 1000,\ h = 1193) = 2000 \text{ lbm/sec-ft}^2$

$W_{crit}/A = (27.94 \text{ lbm/sec}) / (2 \text{ in}^2 / 144 \text{ in}^2/\text{ft}^2)$

$W_{crit}/A = 2012 \text{ lbm/sec-ft}^2$

Checks

Two-phase Critical Flow Using Moody Homogeneous Equilibrium Model (Ref. 1 & 2) (cont)

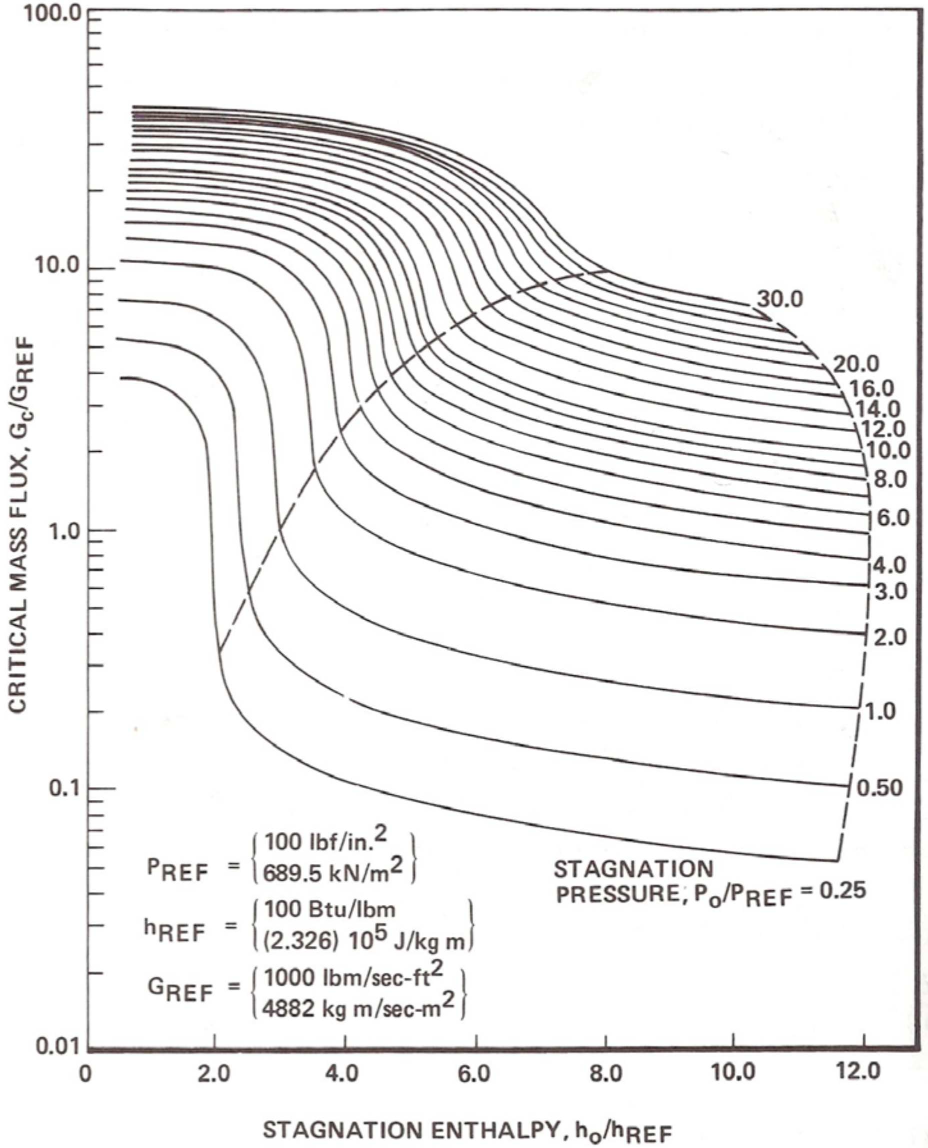


Fig. 9-11a. Critical mass flux-homogeneous, equilibrium steam-water (Moody, 1975).

Two-phase Critical Flow Comparing Moody, HEM and Bernoulli (Refs. 1 & 2) (cont)

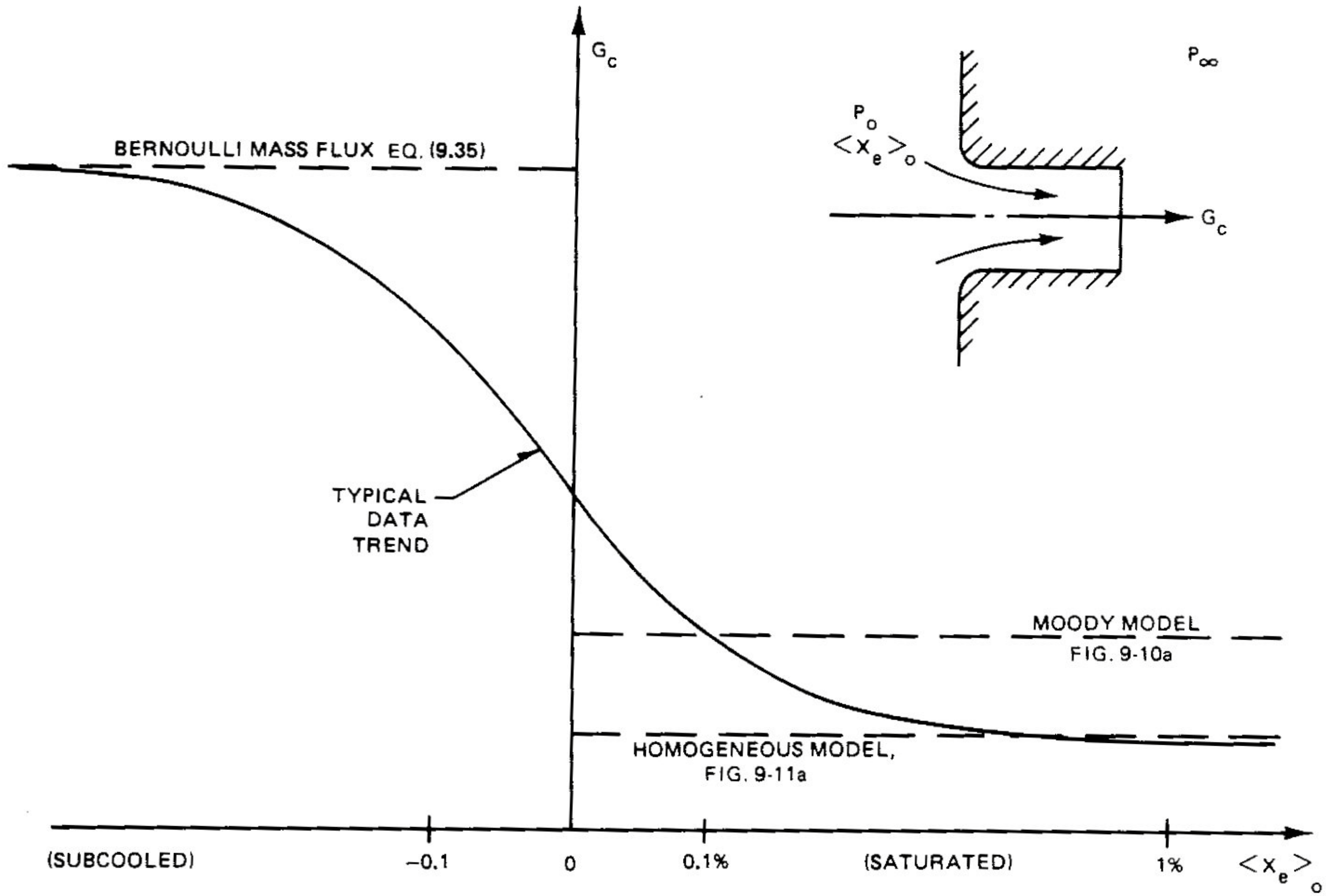
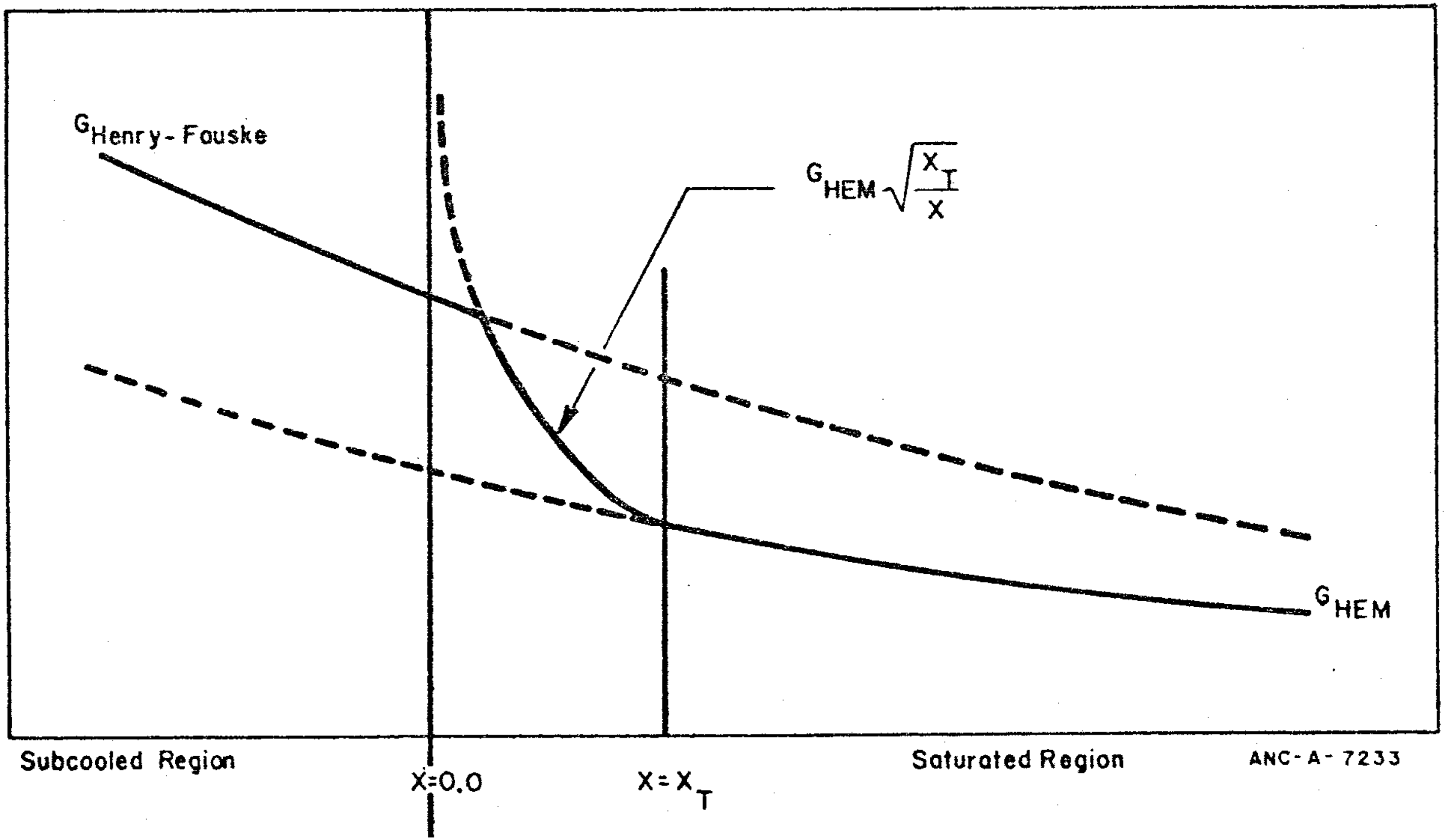


Fig. 9-12. Typical variation of critical mass flux with inlet quality.

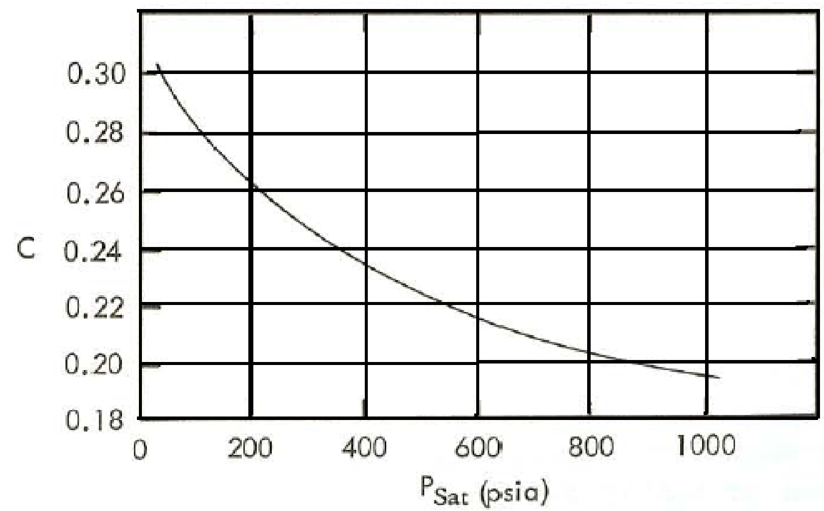
Henry-Fauske/ Homogeneous Equilibrium Critical Flow Transition Model



Brunell Critical Flow for Flashing Water

- As with anything dealing with two phase flow:
- Assumed Slip Ratio affects two-phase critical flow correlations.
- For example: Burnell developed critical flow correlation for flow of flashing water in short pipes and orifices

  $$G_{crit} = (2g_c\rho_l(P - (1\text{-}C)P_{sat}))^{1/2}$$
- Other critical flow correlations exist for different assumed regimes



Pressure coefficient for Burnell critical-flow equation.

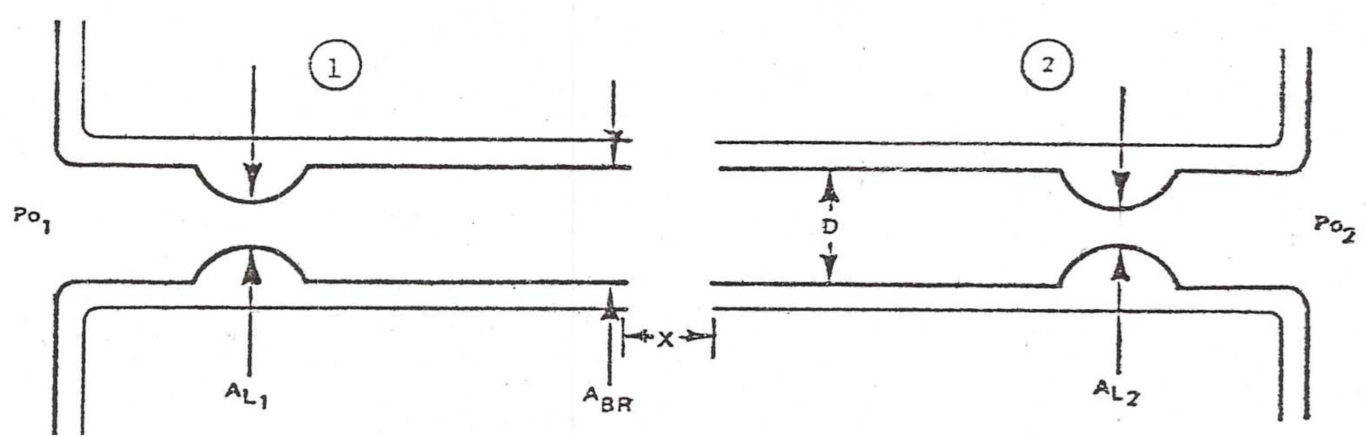
4. Where is Critical Flow Important

- Used in LOCA break flow calculations
- Main Steam Line Breaks
- Maximum flow in a orifice
- Safety Valve Opening

LOCA break flow calculations

- Small Breaks
- Large Breaks
- Intermediate Breaks
- Include Break side and discharge side, especially to determine flow into a containment or sub compartment

Break Flow from Discharge Pipe for double-ended Rupture



Reference: NEDO 24548

ASSUMPTIONS:

- The initial Velocity of the fluid in the pipe is zero. When considering both sides of the break, the effects of initial velocities would tend to cancel out.
- Constant reservoir pressure at 1000 psia at saturated liquid conditions.
- Initially fluid conditions inside the pipe on both sides of the break are similar.
- Wall thickness of the pipe is small compared to the diameter.
- Subcompartment pressure ≈ 0 psig.
- Quasi-steady mass flux is calculated using the Moody steady slip flow model with subcooling.

Nomenclature and Definitions

A_{BR} - Break Area, ft

A_L - Minimum cross-sectional area between the vessel and the break. This is the sum of the area of parallel flow paths.

c_1 - Sonic speed in the fluid

D - Pipe inside diameter at the break location, ft

F_I - Inventory flow multiplier

F_I = 0.75 for saturated steam

F_I = 0.50 for liquid

g_c - Proportionality constant (= 32.17 lbm-ft/lbf-sec^2)

G_1 - Mass Flux, lbm/ft^2 -sec

G_c - Maximum mass flux, lbm/ft^2 -sec

h_o - Reservoir or vessel enthalpy, BTU/lbm

h_p - Initial enthalpy of the fluid in the pipe, BTU/lbm

L_I - Inventory length. The distance between the break and the nearest area decrease of A_L .

$\dot{M}$ - Mass Flowrate, lbm/sec

$\dot{M}_I$ - Mass flowrate during the inventory period, lbm/sec

P_o - Reservoir or vessel pressure, 1000 psia

P_{sat} - Saturation pressure for liquid with an enthalpy of h_p

t - Time, sec

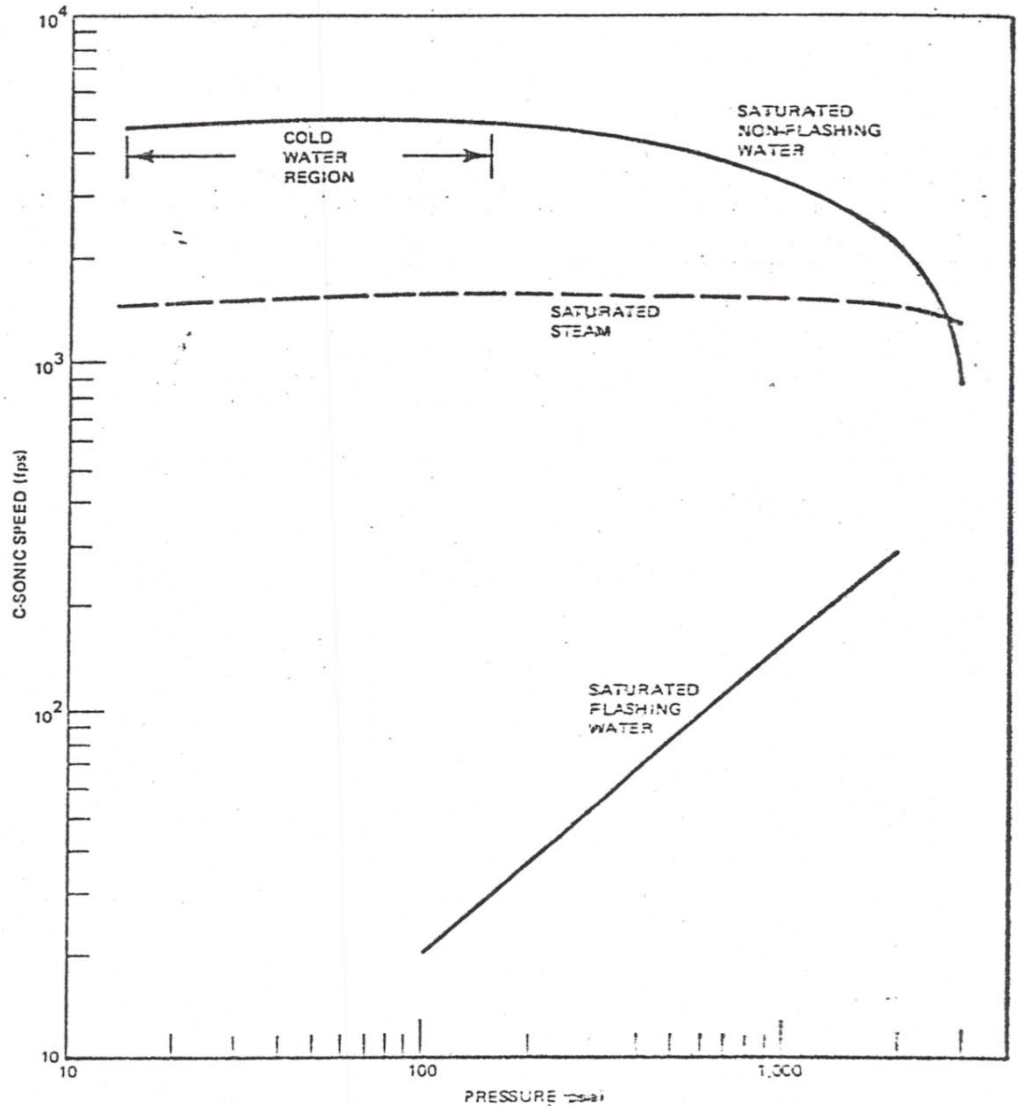
t_I - Length of the inventory period, sec

v - specific volume of the fluid initially in the pipe, ft ^{3}/ lbm

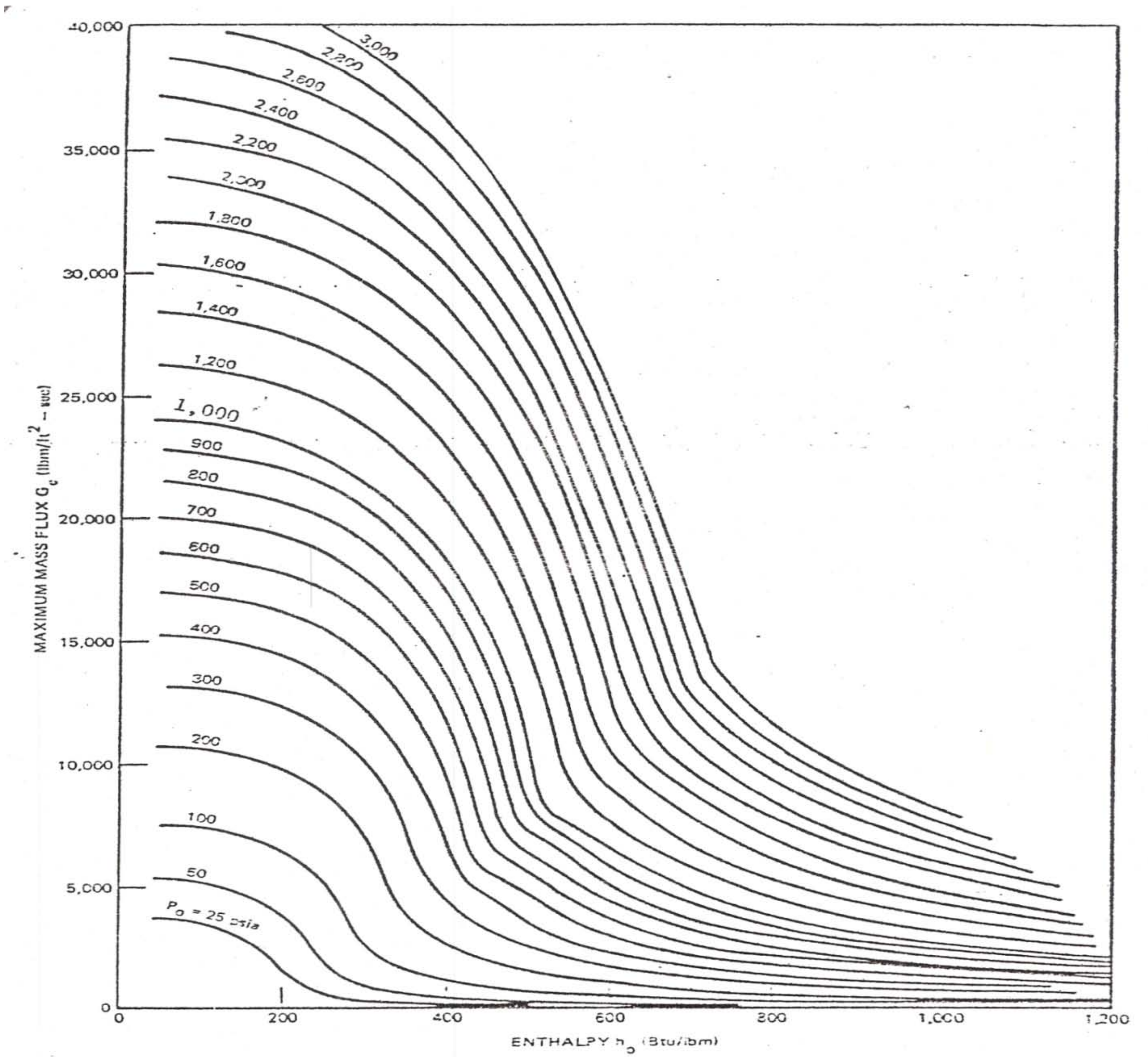
V_I - Volume of the pipe between the break and A_L , ft ^{3}

X - Separation distance of the break, ft

Sonic Speed in Water versus Water Pressure



Moody Equilibrium Critical Flow with Slip Extended into Subcooled Region



Short Term Release Rate Calculations

Short Term Release Rate Calculations

$h_f := 542.6 \quad \frac{Btu}{lbm}$ From ASME Steam Tables

$h_o := h_f$

$F_l := 0.50$ for liquid from NEDO-24548 pg.11

$D := 2.5 \quad ft$ Inside diameter of pipe at break down location (Assumed)

$D_L := 0.5 \quad ft$ Diameter of restriction (Assumed)

$L_l := 26 \quad ft$ The distance between break and nearest restriction

$P_o := 1000 \quad psi$ Initial pressure

Short Term Release Rate Calculations (cont)

$G_c := 7500 \ \frac{\text{lbm}}{\text{ft}^2}$ Maximum mass flux from Moody Plots

$G_1 := G_c$ mass flux from Moody Plots

$c_1 := 1.6 \cdot 10^2 \ \frac{\text{ft}}{\text{s}}$ sonic speed in the fluid (from Sonic Speed in Water vs. Water Pressure Figure

$v := 0.02159 \ \frac{\text{ft}^{3.}}{\text{lbm}}$ specific volume of fluid initially in the pipe from ASME Steam Tables

$A_{BR} := \frac{1}{4}\pi \cdot D^2$ $A_{BR} = 4.909 \ \text{ft}^2$ Break Area

$A_L := \frac{1}{4}\pi \cdot D_L{}^2$ $A_L = 0.196 \ \text{ft}^2$ Mimimum cross sectional area between vessel and break

$V_I := A_{BR} \cdot L_I$ $V_I = 127.627 \ \text{ft}^3$ Volume of the pipe between the break and A_L

Short Term Release Rate Calculations (cont)

$$\frac{A_L}{A_{BR}} = 0.04 \qquad F_I = 0.5$$

If $\dfrac{A_L}{A_{BR}} > F_I$

$$t_I := \frac{(2 \cdot L_I)}{c_1} \qquad t_I = 0.325 \quad \text{sec}$$

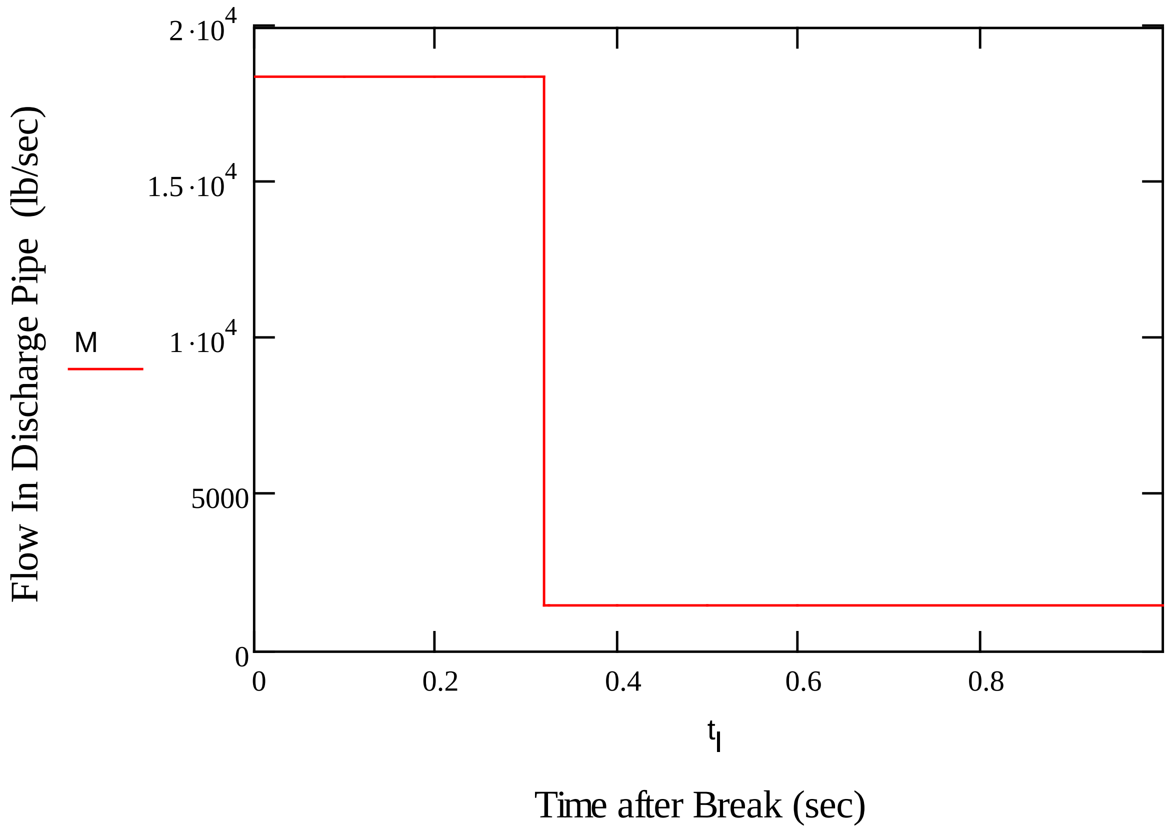
If $\dfrac{A_L}{A_{BR}} < F_I$

$$t_{I.} := \frac{V_I}{A_{BR} \cdot G_1 \cdot F_I \cdot v} \qquad t_{I.} = 0.321 \quad \text{sec}$$

Short Term Release Rate Calculations (cont)

$M_I := G_1 \cdot A_{BR} \cdot F_I$ $\quad$ $M_I = 1.841 \times 10^4 \, \frac{lb}{sec}$ $\quad$ Flow during t_I

$M_{ss} := A_L \cdot G_1$ $\quad$ $M_{ss} = 1.473 \times 10^3 \, \frac{lb}{sec}$ $\quad$ Steady state flow through restriction after t_I



5. Describe Origins and Use of Critical Flow Correlations

NAME OF THE FIRST AUTHOR AND PUBLICATION DATE	MAIN MODEL CHARACTERISTIC	MAXIMUM FLOWRATE EXPRESSION (WHEN ACHIEVABLE)
PERFECT GAS CLASSICAL THEORY	/	$\Gamma = \left[\gamma \, p_o \, \rho_o \left(\frac{2}{\gamma+1}\right)^{\frac{\gamma+1}{\gamma-1}} \right]^{\frac{1}{2}}$
PERFECT GAS (ASSUMPTION i))	/	$\Gamma = \max(p) \left[\rho(p_o - p) \right]^{\frac{1}{2}}$
PERFECT GAS (ASSUMPTION ii))	/	$\Gamma = \max(p) \, \rho \left[2(h_o - h) \right]^{\frac{1}{2}}$
INCOMPRESSIBLE LIQUID	/	$\Gamma = \left[2 \rho (p_o - p_e) \right]^{\frac{1}{2}}$

NAME OF THE FIRST AUTHOR AND PUBLICATION DATE	MAIN MODEL CHARACTERISTIC	MAXIMUM FLOWRATE EXPRESSION (WHEN ACHIEVABLE)
HOMOGENEOUS EQUILIBRIUM MODEL	HEM	$\Gamma = \max(p) \dfrac{2\left[h_o(p_o) - h_g(p) - \dfrac{s_o(p_o) - s_f(p)}{s_{fg}(p)} h_{fg}(p) \right]^{\frac{1}{2}}}{v_f + \dfrac{s_o(p_o) - s_f(p)}{s_{fg}(p)} v_{fg}(p)}$
LAHEY et al. 1977	FLUID-DYNAMIC APPROACH	$\Gamma = \left(\dfrac{-1}{\left\langle \left\{ \dfrac{dv_f}{dp} - \dfrac{v_{fg}}{s_{fg}} \dfrac{ds_g}{dp} + x \left[\dfrac{dv_{fg}}{dp} - \dfrac{v_{fg}}{s_{fg}} \dfrac{ds_{fg}}{dp} \right] \right\} \right\rangle} \right)^{\frac{1}{2}} = f(p_e)$

5. Describe Origins and Use of Critical Flow Correlations (cont.)

NAME OF THE FIRST AUTHOR AND PUBLICATION DATE	MAIN MODEL CHARACTERISTIC	MAXIMUM FLOWRATE EXPRESSION (WHEN ACHIEVABLE)
MOODY 1965	SLIP EQUILIBRIUM MODEL (ENERGY MODEL)	$\Gamma = \max(p,k) \left\{ \dfrac{2\left[h_o - h_f - \dfrac{h_{fg}}{s_{fg}}(s_o - s_f)\right]}{\left[\dfrac{k(s_g - s_o)}{s_{fg}} v_f + \dfrac{s_o - s_f}{s_{fg}} v_g\right]^2 \left[\dfrac{s_o - s_f}{s_{fg}} + \dfrac{s_g - s_o}{k^2 s_{fg}}\right]} \right\}^{\frac{1}{2}}$
FAUSKE 1964	SLIP EQUILIBRIUM MODEL (MOMENTUM MODEL)	$\Gamma = \left\{ \dfrac{-k}{(1-x+kx)x\dfrac{dv_g}{dp} + \left[v_g(1+2xk-2x) + v_f(2xk-2k-2xk^2+k^2)\right]\dfrac{dx}{dp}} \right\}^{\frac{1}{2}}$ $= f(p_e);\ x = f(p_e);\ k = f(p_e)$
CRUVER et al. 1967	DEFINITION AND ANALYSIS: Area average specific volume, momentum average specific volume, kinetic energy average specific volume, velocity weighted specific volume	$\Gamma = \left\{ \dfrac{-v_H}{v_f\left[1+x(k^2-1)\right]^2 \left[\dfrac{x}{k}\left(\dfrac{\partial v_g}{\partial p}\right)_s + \dfrac{3}{2} v_f (k^2-1)\left(\dfrac{\partial x}{\partial p}\right)_s + (1-x)\left(\dfrac{\partial v_f}{\partial p}\right)_s\right]} \right\}^{\frac{1}{2}}$ $ds = 0;\ k = (v_g/v_f)^{1/3}$

5. Describe Origins and Use of Critical Flow Correlations (cont.)

NAME OF THE FIRST AUTHOR AND PUBLICATION DATE	MAIN MODEL CHARACTERISTIC	MAXIMUM FLOWRATE EXPRESSION (WHEN ACHIEVABLE)
CASTIGLIA et al. 1979	MAXIMUM ENTROPIC FLOW	AS MOODY 1965 WITH CONDITIONS $\frac{\partial s}{\partial \alpha} = 0$, $\frac{\partial s}{\partial p} = 0$
WALLIS et al. 1978	ISENTROPIC STREAM TUBE MODEL	$\Gamma = \max(p) \left[\sum_{i=1}^{n} \frac{Y_i}{\rho_{i,n} w_{i,n}} + \frac{Y_n}{\rho_{fn} w_n} \right]^{-1}$
BURNELL 1947	SEMIEMPIRICAL CORRELATION	$\Gamma = \{ 2\rho_f [p_{up} - (1 - C) p_{sat}] \}^{1/2}$
ZALOUDEK 1963	AS ABOVE	$\Gamma = C_1 \{ [2\rho_f (p_{up} - p_{sat})] \}^{1/2}$
STARKMAN et al. 1964	FROZEN COMPOSITION	$\Gamma = \frac{1}{x_o v_g + (1 - x_o) v_{go}} \left[2 x_o v_{go} p_o \frac{\gamma}{\gamma - 1} \left(1 - \left(\frac{p_c}{p_o}\right)^{\frac{\gamma-1}{\gamma}} \right) \right]^{1/2}$
MOODY 1969	PRESSURE PULSE MODEL	$\Gamma^2 = \frac{-\left[\frac{\alpha^2}{x^2} \rho_g + \frac{(1-\alpha)^2}{(1-x)^2} \rho_f\right]}{\frac{\alpha}{v_g}\left(\frac{\partial v_g}{\partial p}\right)_s + \frac{1-\alpha}{v_f}\left(\frac{\partial v_f}{\partial p}\right)_s}$; $x = x_o$; k=fixed

5. Describe Origins and Use of Critical Flow Correlations (cont.)

NAME OF THE FIRST AUTHOR AND PUBLICATION DATE	MAIN MODEL CHARACTERISTIC	MAXIMUM FLOWRATE EXPRESSION (WHEN ACHIEVABLE)
CASTIGLIA et al. 1979	MAXIMUM ENTROPIC FLOW	AS MOODY 1965 WITH CONDITIONS $\frac{\partial s}{\partial \alpha} = 0, \quad \frac{\partial s}{\partial p} = 0$
WALLIS et al. 1978	ISENTROPIC STREAM TUBE MODEL	$\Gamma = \max(p) \left[\sum_{i=1}^{n} \frac{Y_i}{\rho_{i,n} w_{i.n}} + \frac{Y_n}{\rho_{fn} w_n} \right]^{-1}$
BURNELL 1947	SEMIEMPIRICAL CORRELATION	$\Gamma = \{ 2\rho_f [p_{up} - (1 - C) p_{sat}] \}^{1/2}$
ZALOUDEK 1963	AS ABOVE	$\Gamma = C_1 \{ [2\rho_f (p_{up} - p_{sat})] \}^{1/2}$
CASTIGLIA et al. 1979	MAXIMUM ENTROPIC FLOW	AS MOODY 1965 WITH CONDITIONS $\frac{\partial s}{\partial \alpha} = 0, \quad \frac{\partial s}{\partial p} = 0$
WALLIS et al. 1978	ISENTROPIC STREAM TUBE MODEL	$\Gamma = \max(p) \left[\sum_{i=1}^{n} \frac{Y_i}{\rho_{i,n} w_{i.n}} + \frac{Y_n}{\rho_{fn} w_n} \right]^{-1}$
BURNELL 1947	SEMIEMPIRICAL CORRELATION	$\Gamma = \{ 2\rho_f [p_{up} - (1 - C) p_{sat}] \}^{1/2}$
ZALOUDEK 1963	AS ABOVE	$\Gamma = C_1 \{ [2\rho_f (p_{up} - p_{sat})] \}^{1/2}$

5. Describe Origins and Use of Critical Flow Correlations (cont.)

NAME OF THE FIRST AUTHOR AND PUBLICATION DATE	MAIN MODEL CHARACTERISTIC	MAXIMUM FLOWRATE EXPRESSION (WHEN ACHIEVABLE)
ARDRON et al. 1976	UPPER BOUND FLOW	$\Gamma_c^2 = \left[\dot{\rho}_{go} w_{go}\right]^2 \left(\frac{p_c}{p_o}\right)^{\frac{\gamma+1}{\gamma}} \frac{1}{x_o^2} \frac{1+\left(\frac{1-x}{x}\right)\frac{\rho_g}{\rho_f}}{\left[1+\left(\frac{\rho_f}{\rho_{go}}\right)^{1/3}\left(\frac{p_o}{p_c}\right)^{1/3\gamma}\left(\frac{1-x}{x}\frac{\rho_g}{\rho_f}\right)\right]^2}$
HENRY et al. 1970	LOW QUALITIES (x_E <0.02) MODEL	$\Gamma_c = \frac{\Gamma_{cHE}}{\sqrt{N_e}}$
HENRY 1970	HIGH L/D RATIOS (L/D ≈ 12)	$\Gamma_c^2 = -\left[\left(v_g - v_{fo}\right) N \frac{dx_E}{dp}\right]_c^{-1}$
HENRY 1970	VERY HIGH L/D RATIOS (L/D > 12)	$\Gamma_c^2 = \left[\frac{x v_g}{p} - \left(v_g - v_{fo}\right) N \frac{dx_E}{dp}\right]_c^{-1}$

Use of Critical Flow Correlations in Nuclear Applications

- Most common use of HEM, Moody, and Henry Fauske
- Used in LOCA break flow calculations
- Main Steam Line Breaks
- Maximum flow in a orifice
- Safety Valve Opening

Use of Critical Flow Correlations in Nuclear Applications (cont.)

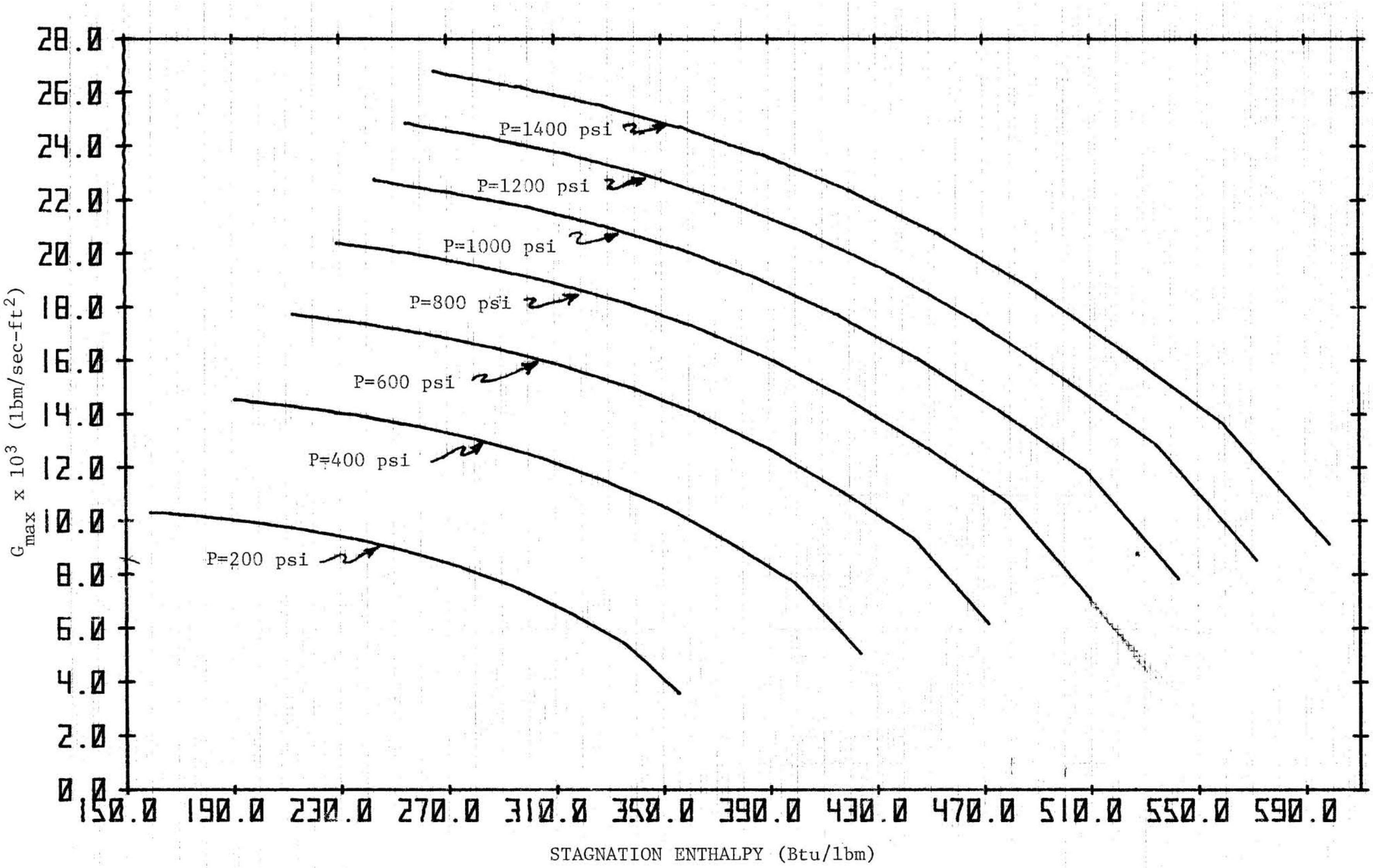
10CFR50 Appendix K--ECCS Evaluation Models

I. Required and Acceptable Features of the Evaluation Models

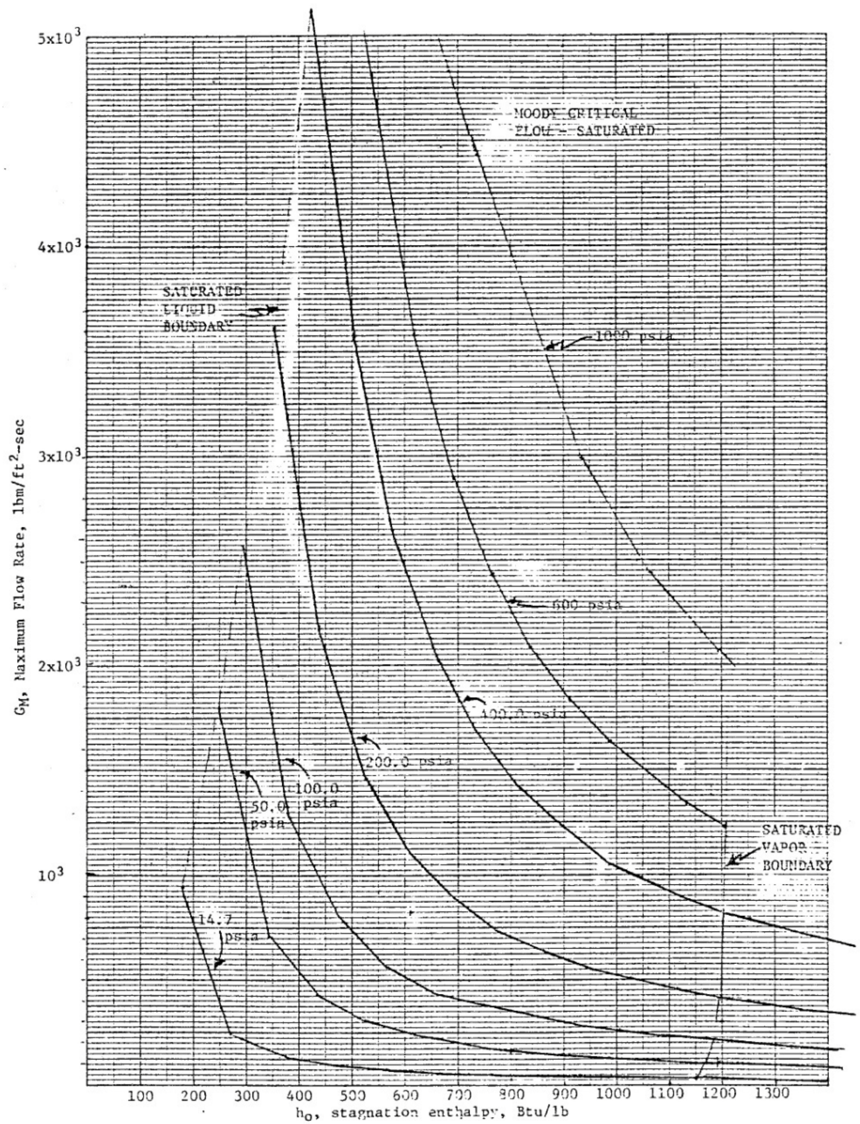
C. Blowdown Phenomena

b. *Discharge Model*. For all times after the discharging fluid has been calculated to be two-phase in composition, the discharge rate shall be calculated by use of the Moody model (F.J. Moody, "Maximum Flow Rate of a Single Component, Two-Phase Mixture," Journal of Heat Transfer, Trans American Society of Mechanical Engineers, 87, No. 1, February, 1965). The calculation shall be conducted with at least three values of a discharge coefficient applied to the postulated break area, these values spanning the range from 0.6 to 1.0. If the results indicate that the maximum clad temperature for the hypothetical accident is to be found at an even lower value of the discharge coefficient, the range of discharge coefficients shall be extended until the maximum clad temperatures calculated by this variation has been achieved.

Henry/Fauske Critical Flow

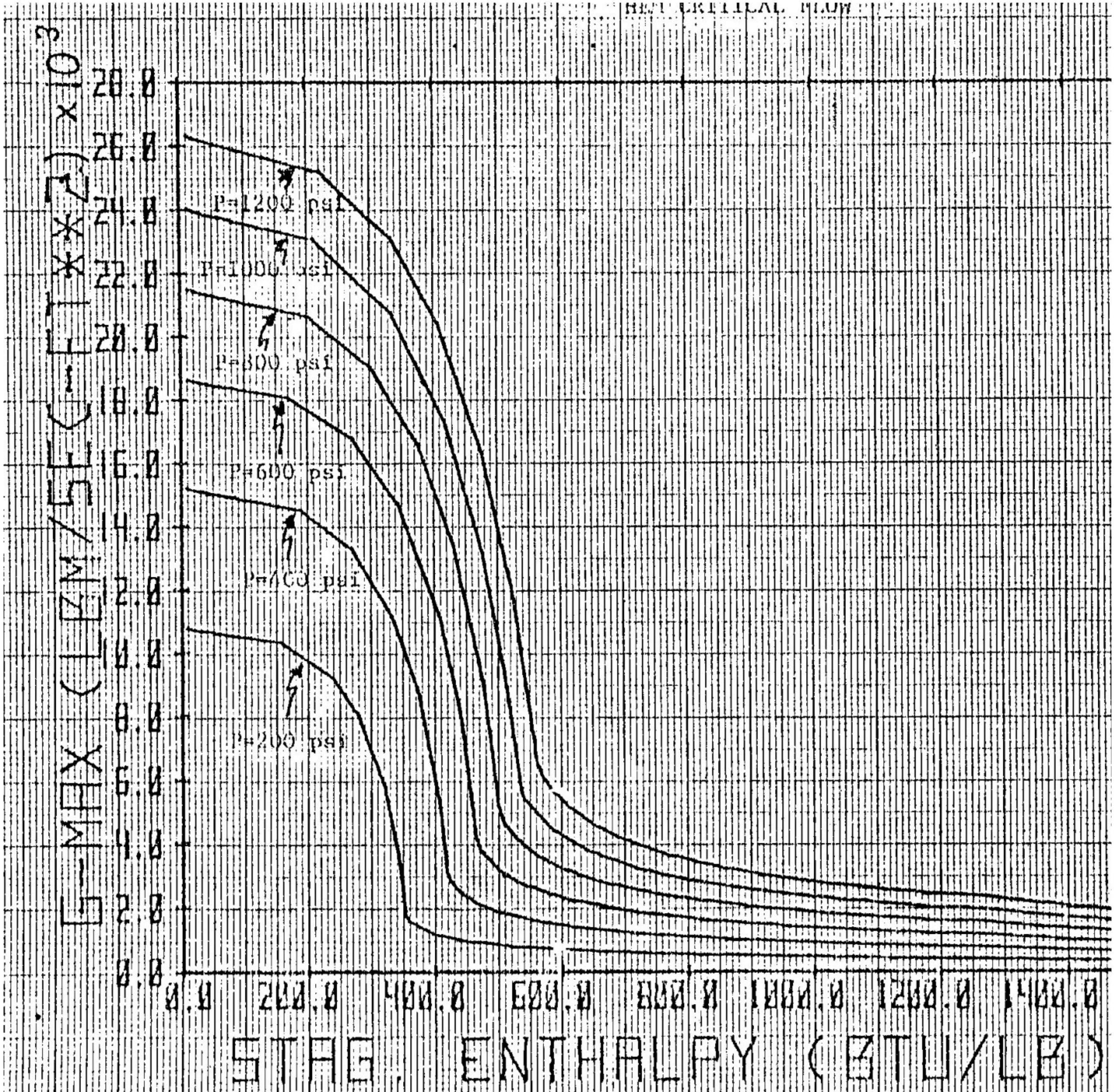


Moody Critical Flow

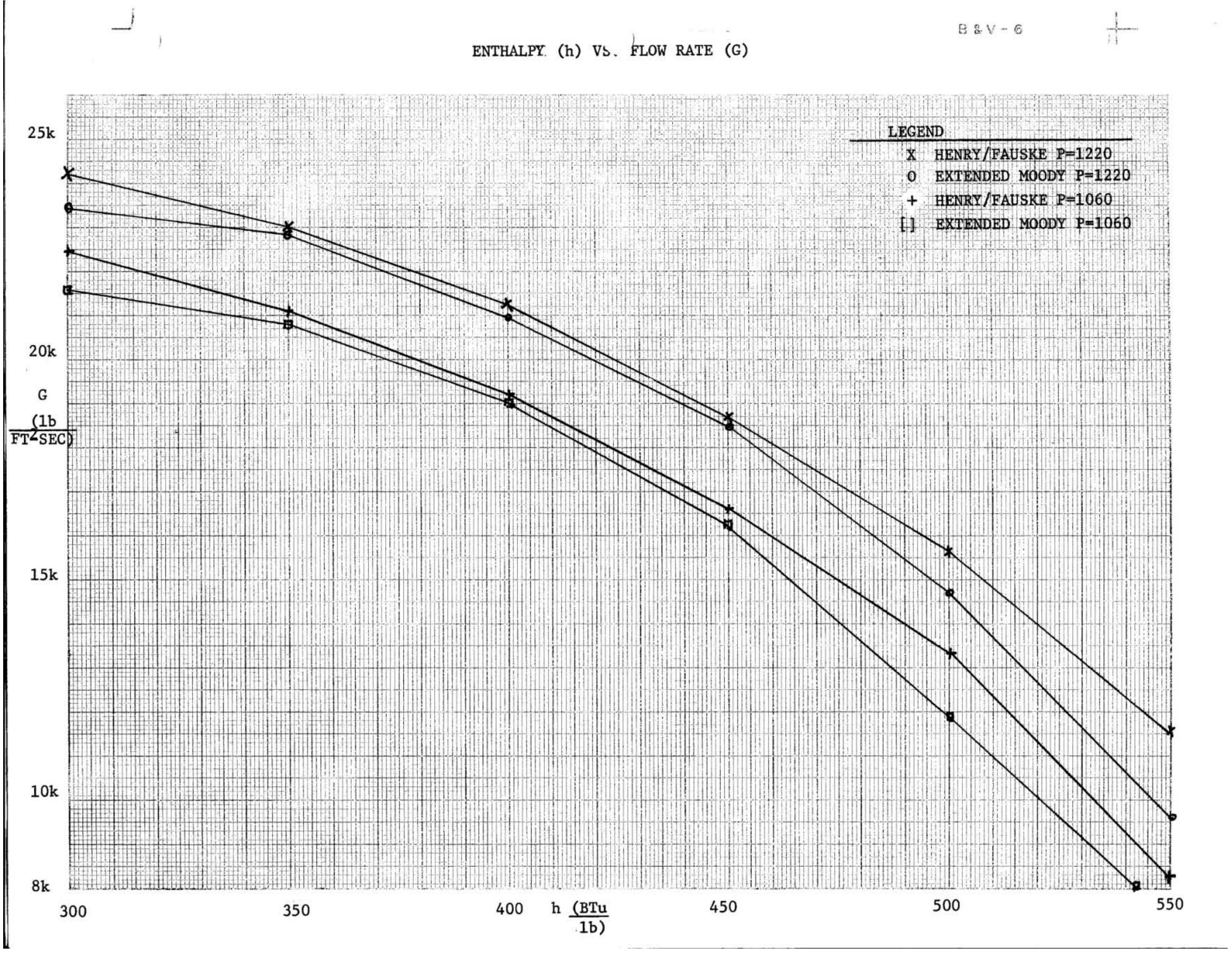


MOODY CRITICAL FLOW IN THE SATURATED REGION PLOTS

HEM Critical Flow



H/F versus Moody Extended Subcooled



6. Describe Some Testing That has Been Performed

- **CSNI Code Validation Matrix** **http://www.nea.fr/html/dbprog/ccvm/index.html**
- **INTEGRAL TEST DATA**
- **Code Validation Matrix of Thermo-Hydraulic Codes for LWR LOCA and Transients**

GENERAL

CSNI Integral Test Facility Validation Matrix for the Assessment of Thermal-Hydraulic Codes for LWR LOCA and Transients
Prepared by a Writing Group Committee of the PWG 2 "Task Group on Thermal-Hydraulic System Behaviour"
OCDE/GD(97)12, NEA/CSNI/R(96)17 (July 1996)

BETHSY

The BETHSY integral test facility located in the Nuclear Research Center in Grenoble (France) is a scaled down model of a 3 loop 900 eMW FRAMATOME PWR; the overall scaling factor applied to every volume, mass flowrate and power level is close to 1/100, while elevations are 1/1 in order to preserve the gravitational heads.

DOEL2

DOEL 2 is a Westinghouse, 2 loop pressurized water reactor (PWR) rated at 392 Mwe (NET) and commissioned in 1975, for which TRACTEBEL was the architect/engineer. This plant located in Belgium in part of the twin concept with DOEL 1, as they share some common engineered safety systems such as the high pressure safety injection system (HPSI). Data measured during the Steam Generator Tube Rupture (SGTR) incident on full scale facility are available.

Describe Some Testing That has Been Performed (cont)

FIST

BWR/6-218 standard plant. A full size bundle with electrically heated rods is used to simulate the reactor core. A scaling ratio of 1/624 is applied in the design of the system components. Key features of the FIST facility include: (1) Full height test vessel and internals; (2) correctly scaled fluid volume distribution; (3) simulation of ECCS, S/RV, and ADS; (4) level trip capability; (5) heated feedwater supply system, which provides the capability for steady state operation.

FIX-II

May 31,1988 report: FIX-II/2032, BWR Pump Trip Experiment 2032, Simulation Mass Flow and Power Transients. In the FIX-II pump trip experiments, mass flow and power transients were simulated subsequent to a total loss of power to the recirculation pumps in an internal pump boiling water reactor. The aim was to determine the initial power limit to give dryout in the fuel bundle for the specified transient. In addition, the peak cladding temperature was measured and the rewetting was studied.

LEIBSTADT

August 17, 1989 report: LEIBSTADT/STP-2001, BWR/6, Reactor Core Isolation Cooling System Test. The Leibstadt Nuclear Power Station is equipped with a direct cycle boiling water reactor belonging to the General Electric BWR product line BWR/6. The MARK III containment system encloses the nuclear island. The nuclear system is provided with a 238-inch internal diameter vessel and the core is built up of 648 fuel elements and 84 control rods. Each fuel bundle consists of 62 fuel rods and 2 water rods in an 8 x 8 array. The rated power of the Leibstadt BWR is 3012 MWt and it is designed for a net power of 942 MWe.

Describe Some Testing That has Been Performed (cont)

LOBI

May 22,1992 report: LOBI/A1-04R, Loop for Blowdown Investigation, PWR Double-Ended Cold-Leg Break Test TEST. The LOBI facility is a 1/700 scale model of a four loop PWR and has two primary loops, the intact loop representing three loops and the broken loop representing one loop of a four-loop PWR. The reactor pressure vessel model contains an electrically heated rod-bundle with 64 rods and a heated length of 3.9 m. The nominal heating power is 5.3 MW. The downcomer is of annular shape. An upper head simulator is connected to the vessel. Each of the two primary loops contains a pump and a steam generator. The different mass flows in the loops are established by the pump speeds, since the two pumps are identical. Heat is removed from the steam generators by a secondary system. ECC water can be supplied from two accumulators, one for each loop. Cold or hot leg as well as combined injection can be simulated.

The LOBI test facility is the only high pressure integral test facility within the European Communities (1982), built and operated in the Joint Research Centre, Ispra, ITALY. In the LOBI experiments the thermal-hydraulic behavior of the facility during a transient, caused by a simulated loss of coolant accident (LOCA), is investigated to provide an experimental basis for analytical model development and code verification.

LOFT

May 26,1992 report: The LOFT Integral Test Facility is a scale model of a LPWR. The intent of the facility is to model the nuclear, thermal-hydraulic phenomena which would take place in a LPWR during a LOCA. The general philosophy in scaling coolant volumes and flow areas in LOFT was to use the ratio of the LOFT core [50 MW(t)] to a typical LPWR core [3000 MW(t)]. For some components, this factor is not applied; however, it is used as extensively as practical. In general, components used in LOFT are similar in design to those of a LPWR. Because of scaling and component design, the LOFT LOCA is expected to closely model a LPWR LOCA.

Describe Some Testing That has Been Performed (cont)

OTIS

May 22,1992 report: OTIS is a 1:1686 volume-scaled single loop (one hot log and one cold leg) facility with critical elevations preserved. The single once-through steam generator contains 19 tubes prototypical of a B&W PWR plant. The electrically heated core has a power capability of 180 KW which is representative of 10 per cent scaled power based on a 2584 MWth plant. It contains no primary pump but all other active components are simulated and it operates at full PWR system pressure. The facility tests natural circulation in 1-phase flow; boiler condenser mode; leak flow; heat transfer in covered core; non- condensable gas effects; intermittent two-phase natural circulation; natural circulation in core, vent valve, downcomer; superheating in secondary side.

PACTEL

Nov 26,1998 report: The main goal was to study natural circulation in a VVER plant including several single- and two-phase natural circulation modes. The heat generated in the core is transported by the coolant to the steam generators. The steam generator secondary side conditions were held constant, eliminating disturbances caused by the secondary pressure variation. The primary coolant mass was reduced stepwise in the liquid form, and the amount of the drained water was given as a boundary condition. The draining period is very short compared to the stabilizing period, therefore the different natural circulation mechanisms are clearly identified. It was expected that the characteristics of natural circulation at constant pressure is mostly dependent on the primary coolant inventory and the core power level. During the whole experiment the steam generator level control was maintained. The secondary pressure was controlled by a PI controller.

PIPER

Apr 13,1989 report: The PIPER-ONE facility simulates a General Electric BWR-6 plant. It is characterized by volume and height scaling ratios of 1/2200 and 1/1, respectively. The available core rod electrical power (20% of the nominal value) is sufficient to simulate the nuclear heat decay. No circulation loops are included in the facility, considering their low importance in a small break LOCA and the willingness to achieve the maximum simplicity of the loop operation.

Describe Some Testing That has Been Performed (cont)

PKL

Apr 01,1981 report: PKL-facility simulates the essential primary system components of a typical West German 1300 PWR with regard to their thermohydraulic behaviour. The facility essentially consists of the pressure vessel with the heated bundle, the downcomer simulator, the primary loops with the components steam generator and pump simulator, the injection devices, the break geometry simulator, as well as the separators connected thereto, and the test containment to maintain a back-pressure at the location of break which is expected to be typical for emergency conditions. The number of heater rods and the cross-sections of the testing plant are on a reduced scale 1:134 in comparison with a typical German PWR. The elevations and locations are essentially full scale. Pressure vessel: The space between the pressure vessel and the inner core casing is sealed from the core region and the upper and lower plenum and connected with the upper plenum only by a pressure equalization line. The rod bundle surrounded by the inner core casing consists of 340 rods, 337 of which are indirect electrically heated. The test bundle cross-section as well as a heater element with the measuring elevations, the original-KWU-spacers and the axial power profile (7 power steps) are described.

ROSA-III

June 29,1992 report: This is one of the small break LOCA/ECC test series to study the response of a BWR with ECC injection. Run 912 is a 5% split break test at the recirculation pump inlet. The test is initiated with the steam dome pressure at 7.30 MPa, the lower plenum subcooling at 10.8 K, a core inlet flow rate of 16.4 kg/s, and a core heat generation rate of 3.9 MW. The core is quenched after ECCS actuation and at a maximum fuel cladding temperature of 839 K.

ROSA-IV

June 29,1992 report: ROSA-IV is a large scale test facility (LSTF) for integral simulation of thermal-hydraulic response of a pressurized water reactor during a small break loss-of-coolant accident or an operational transient. The facility has electric core heating. The overall scaling factor is 1/48. The hot and cold legs were sized to conserve the volume scaling. The four primary loops of the reference PWR are represented by two equal-volume loops.

Describe Some Testing That has Been Performed (cont)

SEMISCALE

May 1,1987 report: The Semiscale Mod-3B system is scaled to a reference four-loop PWR. The scaling criterion is a Modified volume scaling based on the ratio of Semiscale power to the thermal power of the reference plant (Trojan). This Scaling produces a scale factor of 1/1705.5. The system consists of a pressure vessel with an electrically-heated, 25-rod PWR core simulator and internals, an external pipe downcomer, and two primary coolant loops. Each loop has an active tube and shell steam generator. The intact loop is scaled to represent three of the four primary loops in a PWR, while the broken loop represents the fourth. Even though S-PL-3 does not incorporate a break, this loop is referred to as the "broken loop". In order to correctly scale the facility and preserve important phenomena, component elevations, dynamic pressure heads, and liquid distributions are maintained as close to the reference PWR values as possible.

SPES

Apr 1,2003 report: The SPES (Simulatore PWR per Esperienze di Sicurezza) integral test facility is a three loop scaled-down model of PWR (Westinghouse 3122 type, 3 loops, 2775 MWth core power) designed for thermal-hydraulic safety research program. SPES test program provides experimental data for the development and assessment of system codes used in PWR safety analysis. SPES is an experimental facility which allows a true simulation of a PWR system as close as possible to the characteristic of the Westinghouse 312 type. The test plant reproduces the primary loops, the most important components and the power channel of the simulated reactor according to significant scaling criteria. The SPES experimental facility, having a 1:427 power-scaling ratio includes a full-length-scale electrical heated power channel and three complete primary loops.

TBL

Apr 21,1989 report: Two Bundle Loop facility. Volume scaling ratio = 1:328. Height ratio = 1:1. The facilities include the following to simulate a BWR system: pressure vessel, main steam system, feedwater, recirculation, ECCS, ADS, break line, utility system, power supply and instrumentation system. The pressure vessel has two heated simulated fuel assemblies.

Describe Some Testing That has Been Performed (cont)

TLTA

May 22,1992 report: The Two-Loop-Test-Apparatus (TLTA) is a 1:624 volume scaled BWR/6 simulator. It was the predecessor of the better-scaled FIST facility. The facility is capable of full BWR system pressure and has a simulated core with a full size 8 x 8, full power single bundle of indirect electrically heated rods. All major BWR systems are simulated including lower plenum, guide tube, core region (bundle and bypass), upper plenum, steam separator, steam dome, annular downcomer, recirculation loops and ECC injection systems. The fundamental scaling consideration was to achieve real-time response. A number of the scaling compromises present in TLTA were corrected in the FIST configuration. These compromises include a number of regional volumes and component elevations.

MARVIKEN

Dec 1979 report: Marviken Full Scale Critical Flow Tests. The major components of the facility are:

The pressure vessel with a net volume = 425 m3, a maximum design pressure = 5.75 MPa and a maximum design temperature = 545 K.

- The discharge pipe attached to the pressure vessel bottom. The discharge pipe consists of a ball valve and pipe spools which house the instrumentation upstream of the test nozzle.
- The test nozzles and rupture disc assemblies. A set of test nozzles of specified lengths and diameters (tubular section lengths from 0.166 to 1.809 m and tubular section diameters from 0.2 to 0.509 m) were used to which the rupture disc assemblies were attached.
- The containment and exhaust pipes, consisting of the drywell, the wetwell, fuel element transport hall, the ground level exhaust pipe, and the upper exhaust pipe.

Example of Test Reports: Semiscale Testing

Summary Report Semiscale Mod-2A Heat Loss Characterization Test Series (EGG-SEMI-5448)

SEMI_R1.pdf Semiscale Mod-1 Program and System Description for the Blowdown Heat Transfer Tests (ANCR-1230)

SEMI_R2.pdf An Evaluation of Piping Heat Transfer Piping Flow Regimes, and Steam Generator Heat Transfer for the Semiscale Mod-1 Isothermal Tests (ANCR-1229)

SEMI_R3.pdf RELAP5 Standard Model Description for the Semiscale Mod-2A System (EGG-SEMI-5692)

SEMI_R4.pdf Steady State Control Model for the Standard RELAP5 Semiscale Mod-2A System Model (RE-A-82-023)

SEMI_R5.pdf Summary of the Semiscale Program (1965-1986) (NUREG/CR-4945-EGG-2509)

SEMI_R6.pdf Semiscale Loss-of-Power Test Results 11th Water Reactor Safety Research Information Meeting, October 24-28, 1983 (ECC-M-22283 Preprint)

SEMI_R7.pdf Semiscale Program Description (TREE-NUREG-1210)

Summary

- Critical Flow is used often in determining maximum flow through breaks and safety valves.
- HEM, Burnell, Bernoulli, Henry-Fauske and Moody are significant critical flow correlations used in the nuclear power industry.
- Many tests have been conducted to validate the critical flows that are used for safety analysis of nuclear power plants.

Selected References

1. R.T. Lahey, Jr. And F.J. Moody, 1977. The Thermal-Hydraulics of a Boiling Water Nuclear Reactor, ANS , LaGrange Park, Ill.
2. Moody, F.J., Maximum Flow Rate of a Single Component, Two-Phase Mixture, ASME Paper 64-HT-35 (1964).
3. Moody, F.J., Maximum Flow Rate of a Single Component, Two-Phase Mixture, J. Heat Transfer, Trans ASME, Ser. C, 87, 134 (1965).
4. B. Chexal, et al. 1991. The Chexal–Lellouche void fraction correlation for generalized application, EPRI NSAC-139.

Selected References (Cont)

5. R.E. Henry, H.F. Fauske, The two-phase critical flow of one component mixtures in nozzles, orifices and short pipes, ASME J. Heat Transfer 93 (1971).
6. F. D'Auria and P. Vigi, Two-Phase Flow Models – A technical Addendum to the CSNI State of the Art Report on Critical Flow Modeling, CSNI Report No. 49, Nuclear Energy Agency, Comitato Nazionale Per L'Energia Nucleare, Rome, (May 1980)
7. F.R. Zaloudek, The Critical Flow of Hot Water Through Short Tubes, HW-77594, General Electric, Hartford Atomic Products Operation, Richland, Washington, (May 1, 1963).

Selected References (Cont)

8. G.S. Lellouche, B.A. Zolotar, Mechanistic model for predicting two-phase void fraction for water in vertical tubes, channels and rod bundles, EPRI NP 2246-SR (1982).
9. F.J. Moody, Maximum two-phase vessel blowdown from pipes, ASME J. Heat Transfer (1966).
10. F.J. Moody, A pressure pulse model for two-phase critical flow and sonic velocity, J. Heat Transfer, (August 1969).
11. Wallis, G.B., One Dimensional Two-Phase Flow. Mc-Graw Hill, New York. (1969)

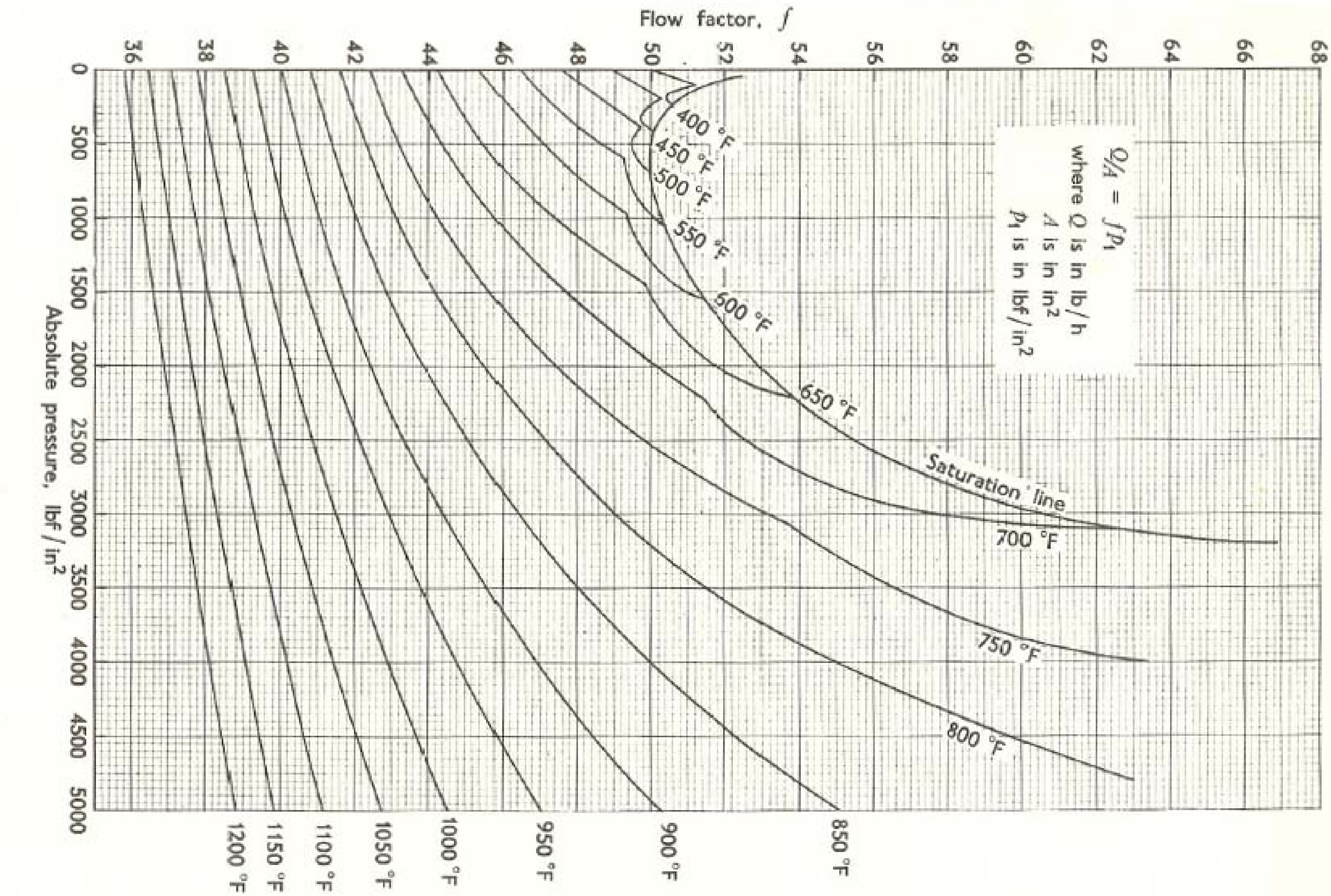


Fig. 5. Critical mass flow per unit area as a function of initial pressure and temperature.

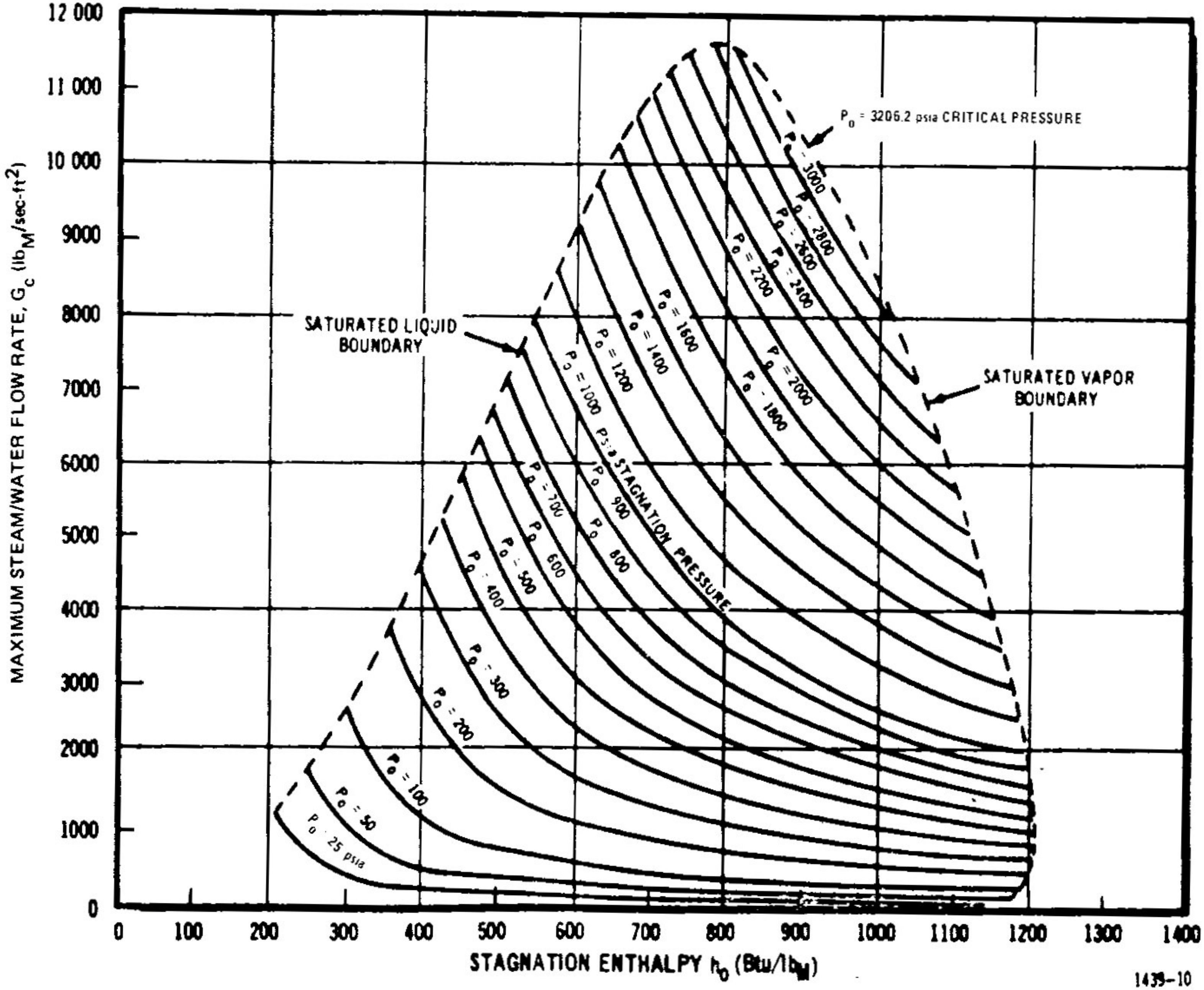
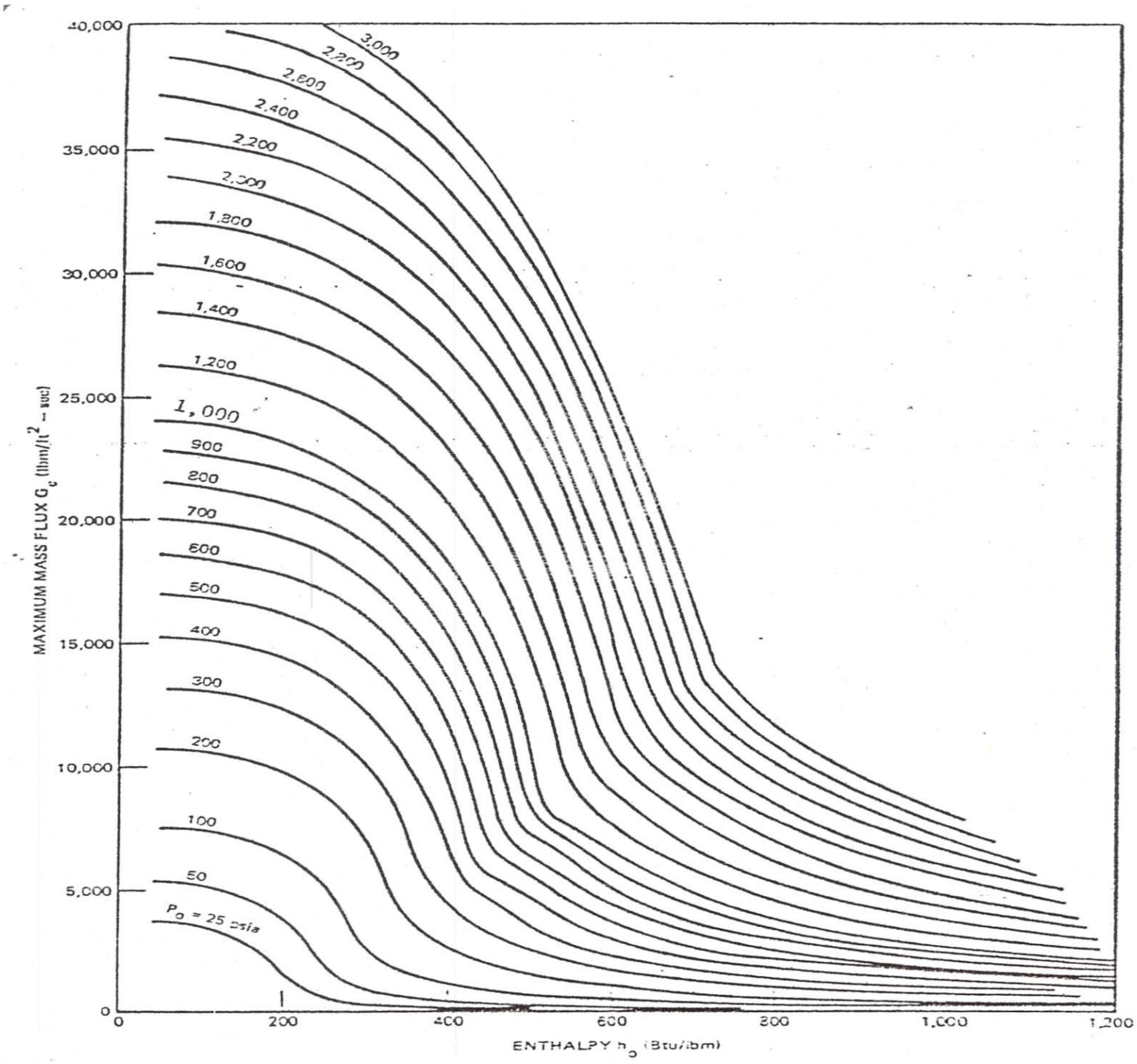
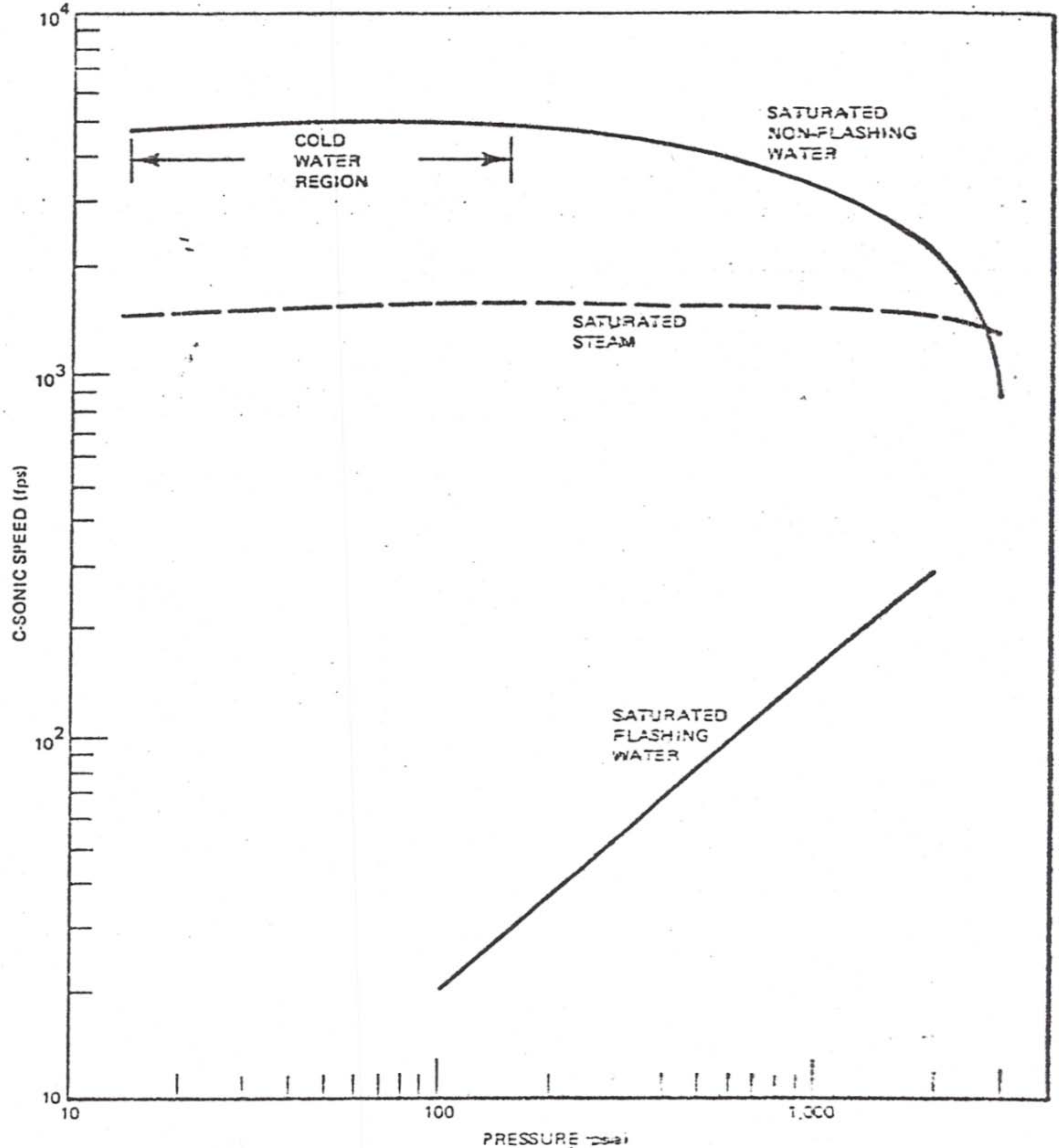


Fig. 9-10a. Maximum steam/water flow rate and local stagnation properties (Moody model).



MAXIMUM MASS FLUX G_c (lbm/ft^2 – sec)
ENTHALPY h_o (Btu/lbm)
P_o = 25 psia
50
100
200
300
400
500
600
700
800
900
1,000
1,200
1,400
1,600
1,800
2,000
2,200
2,400
2,600
2,800
3,000
0
5,000
10,000
15,000
20,000
25,000
30,000
35,000
40,000
200
400
600
800
1,000
1,200



SATURATED
NON-FLASHING
WATER
COLD
WATER
REGION
SATURATED
STEAM
SATURATED
FLASHING
WATER
C-SONIC SPEED (fps)
PRESSURE (psia)
10
100
1,000
10^4
10^3
10^2
10